M-32

PDR/LPDR

Department of Energy
Idaho Operations Office
West Valley Project Office
P.O. Box 191
West Valley, NY 14171

July 23, 1991

Mr. R. Davis Hurt
U. S. Nuclear Regulatory Commission
Headquarters
Washington, D. C. 20555

SUBJECT: Responses to NRC Comments and Actions from the Interface Meeting of June 3-6, 1991

REFERENCES:

1. Letter, J. C. Cwynar to T. J. Rowland, WD:91:C580, "Response to Action Items from NRC Review Meeting of April 2-4, 1991," dated May 30, 1991

2. Letter, P. S. Klanian to T. J. Rowland, WD:91:0596, "Response to NRC Comments on Cement Recipe," dated May 31, 1991

3. Insert for the Waste Qualification Notebook - WVNS-TP-025, Rev. 0

4. Letter, J. L. Mahoney to T. J. Rowland, WD:91:0725, "Meeting Minutes for Cement Waste Form Discussion, NRC Site Visit of June 6, 1991," dated July 5, 1991

5. Selected Well Level and Precipitation/ Temperature Data requested by NRC (Tom Nicolsen)

Dear Mr. Hurt:

This transmittal officially provides to you documentation and data discussed with NRC representatives during both the subject meeting and an earlier interface with your Region I Office on May 16-17, 1991.

Reference 1 provides information on the Evaporator Acid Flush effort, including "split sample" scope which takes into account statistical significance; a tentative schedule for flushing and sampling; and finally information on the accountability of the fissile material in the system.

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Reference 2 provides resolution of NRC comments received on Cement Recipe Qualification in your transmittal of April 22, 1991. The West Valley Nuclear Services Company, Inc. (WVNS) responses were reviewed with Mary Adams during the June 6 meeting with an informal copy provided.

Reference 3, WVNS-TP-025 (4 copies), is an insert to the Waste Form Qualification "notebook" provided to you by transmittal dated May 14, 1991. A copy was provided to Mary Adams during the subject meeting for convenience.

Reference 4 is our version of the "meeting minutes" of the Thursday, June 6 discussions on cement qualification. The minutes reflect some basic agreements on handling additional information as it is generated. For example, the Test Plan for cement qualification "changes" as time marches on. WVNS documents these changes with a "test exception" control system. During the meeting we discussed the issue of how frequently these changes should be passed on to NRC. Our agreement with Mary Adams and Gary Comfort was that we would incorporate the changes into revised Test Plan documentation as appropriate and only forward approved final revisions to you. This will allow us to conduct our activities in a "normal" fashion, and will prevent sending information that may seem somewhat insignificant upon receipt.

The minutes also reflect agreement on earlier discussions between NRC and West Valley on the withdrawal of the proposed "Topical Report" for Cement Qualification. All parties are in agreement that the TSRs (Test Summary Reports) as described in the Waste Qualification Notebook, will provide the same information as is conventionally expected in a Topical. This letter documents our mutual agreement and authorizes WVNS to delete the Topical Report deliverable. We will proceed to remove this activity from the schedule, which will be reflected in one of the upcoming schedule updates.

Reference 5 is some additional data on monitoring wells and weather conditions, requested during the presentation of the NDA Groundwater Model. While we applaud the NRC efforts to develop a good scientific groundwater model, serious concerns over the applicability of the model to the "normal" conditions experienced in the NDA enclosure were expressed during the meeting. The concerns are based primarily on the data against which the model was correlated - most of it was collected during the construction phases of the NDA Interceptor Trench project. This represents a highly unusual set of conditions which would introduce significant amounts of non-typical information upon which the model is based. Hopefully, the enclosed data will help this situation somewhat. Now that the trench is completed, it should continue to stabilize over time, and should allow the groundwater

and run-off to approach more "typical" behavior. However, it would be nice to have much more data (say 3-5 years worth) before the model could be expected to provide valid results. As indicated in the presentation, NRC hopes to publish this model in an upcoming NUREG. We would like to request an opportunity to review the "draft" before it is published, to assist in providing a reasonably accurate model.

Should you have any questions, please call A. Yeazel on (716) 942-4780 or Steve Ketola (716) 942-4314.

Sincerely,

Tom Rowland

T. J. Rowland, Director
West Valley Project Office

Enclosure

cc: P. S. Klanian, WVNS
P. C. Newsom, WVNS

WSK:020:91 - 0894:91:08
0700:91:11
0254:91:10

WSK/sl

West Valley
Nuclear Services Company
Incorporated

WD:91:0580

P.O. Box 191
West Valley, New York 14171-0191
MS-W

RESPONSES:
DW:4379
DW:4382
DW:4383

May 30, 1991

Mr. T. J. Rowland, Director
West Valley Project Office
U. S. Department of Energy
MS-DOE
P. O. Box 191
West Valley, New York 14171-0191

Dear Mr. Rowland:

Attention: J. A. Yeazel

SUBJECT: Response to Action Items from NRC Review Meeting of April 2-4, 1991

References: 1) WD:91:0414, "NRC Requests per April 4, 1991 Closeout Meeting", M. N. Haas to T. J. Rowland dated April 19, 1991.

2) Standard Operating Procedure SOP 15-4, "Sample Operations", latest revision.

The purpose of this letter is to provide responses to three of the action items due by May 30, 1991 resulting from the Nuclear Regulatory Commission (NRC) visit of April 2-4, 1991, during which plans for resumption of waste tank farm processing were reviewed. The complete list of action items is documented in reference 1. The action items to be addressed by this letter are reiterated below along with the West Valley Nuclear Services (WVNS) response.

ACTION NO. 1 **Split Samples**

a. of evaporator acid flush (quantity/frequency to be determined)

Based on discussions with J. Roth of the NRC on May 16, 1991, the following sampling protocol was agreed to:

1. The spent evaporator acid cleaning solution will be sampled in concentrates tank 5D-15A2 after cooling to ambient cell temperature in accordance with the normal WVNS sampling procedure SOP 15-4 (reference 2). This includes an air sparge prior to and during sampling to assure homogeneity.

2. The number of samples shall be in accordance with a statistically derived sample schedule developed by WVNS (see attachment A). A total of 35 samples is recommended with the following disposition:

SRC4112

0780:91:11

Ship to NRC for immediate analysis	4
Analytical and Process Chemistry (A&PC) analyze immediately	4
Hold for future A&PC analyses	10
Hold for future NRC analyses	10
Hold as archive	7
Total	35

3. The sample volume shall be approximately 10 ml.

4. WVNS (G. A. Smith/J. P. Jackson) will package and ship NRC samples to:

   Robert Oldham
   New Brunswick Lab
   Building D-350
   USDOE
   9800 South Cass Avenue
   Argonne, Illinois 60439-4899

5. The New Brunswick Lab will analyze the NRC samples for total Pu, Pu isotopes, total U, and U isotopes.

6. The A&PC lab will analyze the WVNS samples for total Pu, Pu isotopes, and total U. A&PC lab does not have analytical capability for U isotopes. Consideration is being given to contract with an off-site lab for these analyses. This will be resolved by June 30, 1991 (new commitment).

ACTION NO. 4. Material Balance

Prior to resumption of operation of the IRTS, WVNS needs to quantitatively reconcile the amount of fissile material downstream of the Supernatant Treatment System

WVNS has completed a best estimate of the amount of total plutonium deposited in the LWTS evaporator based on gross alpha activity data and specific gravity measurements for the evaporator feed and product tanks obtained during IRTS Operations (campaigns 1 through 21). The evaluation includes an analysis of the uncertainty in the calculations. It was concluded that 359±100 grams (two sigma) of total plutonium is present in the evaporator. The fissile portion of the total plutonium (Pu-239 and Pu-241) is computed by dividing the total grams of plutonium by 1.23 (aged to 6/1/91) to arrive at 292±81 grams (2 sigma). Details of the calculation and the uncertainty analysis can be found in attachment B.

A similar estimate for the amount of uranium deposited in the evaporator could not be done because uranium analyses were not routinely performed during operation of the evaporator. A bounding estimate has been prepared using recent analytical results from the 8D-2 waste tank and the waste dispensing vessel.

The conservative assumptions of a linear increase of 300 percent in the supernatant from the original 1986 value, coupled with a loss of 41 percent (assumed from day 1 of use) in the evaporator, leads to the bounding estimate of 1886 grams of U-233/235.

ACTION NO. 5. Acid Flush Schedule

The NRC would like advanced notice of when the acid flush is to take place in order to have the opportunity to be present as well as an information copy of the LWTS-SAR review.

The NRC will be notified one week in advance of the start of the acid flush of the evaporator which is currently forecast for the week of June 17, 1991. An information copy of the letter requesting approval of LWTS SAR-005, Rev. 2 is included as attachment C for transmittal to the NRC. This letter also includes resolution of DOE comments.

This letter satisfies commitments DW:4379, DW:4382, and DW:4383. If you have any further questions or comments relative to the responses provided, please contact P. S. Klanian at extension 4382 or J. C. Cwynar at extension 4283.

Very truly yours,

J. C. Cwynar

J. C. Cwynar, Manager
IRTS Process Control Engineering
West Valley Nuclear Services Co., Inc.

DC:91:0055

JCC:src

Attachments:

A) Letter CJ:91:0044, "LWTS Evaporator Cleaning - Sampling Schedule", J. L. Mahoney to P. J. Valenti dated May 22, 1991.

B) Letter CJ:91:0047, "Best Estimate and Uncertainty of Pu in the LWTS Evaporator", J. L. Mahoney to P. J. Valenti dated May 28, 1991.

C) Letter WD:91:0555 (FA:91:0049, RS:91:0024), "Request for Approval of LWTS SAR-005, Rev. 2, Draft C", J. L. Knabenschuh to T. J. Rowland dated May 24, 1991.

cc: W. S. Ketola, DOE-WV Project Office, MS-DOE
C. B. Leek, DOE-WV Project Office, MS-DOE
R. B. Provencher, DOE-WV Project Office, MS-DOE
J. A. Yeazel, DOE-WV Project Office, MS-DOE

SRC4112

From IRTS Engineering

Letter # CJ:91:0044

Date May 22, 1991

Subject LWTS Evaporator Cleaning - Sampling Schedule

To P. J. Valenti MS-W

CC J. C. Crynak MS-W

R. F. Gessner	MS-201	D. C. Meess	MS-Z26
P. S. Klanian	MS-I	MRC (original)	MS-MRC
R. Keel	MS-Z-26	R. A. Humphrey	MS-42
		D. K. Ploetz	MS-305

This memo documents a sampling schedule for spent cleaning solutions from the LWTS evaporator. The schedule is based on standard deviations demonstrated in Waste Dispensing Vessel samples that exhibited a small quantity of solids. A total of 35 samples need to be taken. Four of the samples can be shipped immediately to the NRC for analyses. Four other samples shall be immediately analyzed by the WVNS laboratory. If the percent uncertainty of the mean at the 90% confidence level (equals the standard deviation divided by the square root of the number of samples analyzed times a t statistic) is not less than 10% of the average analysis, then additional samples from the remaining 27 shall be analyzed.

DISCUSSION

Once the acid cleaning of the LWTS evaporator has started, samples of the cooled cleaning solution will be taken from the downstream tank 5D-15A1/A2. The samples will provide the data to quantify the amount of plutonium and uranium removed from the evaporator.

If the cleaning solution is a perfect salt solution only a few samples will be needed to adequately determine the plutonium and uranium content. If solids are present then typically many more samples are needed to reduce the uncertainty in the fissile content in the leaning solution.

0001JLM

During the investigation into the evaporator fissile accumulation, two samples of the Waste Dispensing Vessel were taken. A small amount of solids were detected. On analysis, strontium and plutonium were found to be tied up in the solids. This was confirmed by filtration. Before filtration, the two samples exhibited a standard deviation of about 30% relative. In a sample with solids, this is typical.

Enough samples of the evaporator cleaning solution need to be taken to be certain of the plutonium and uranium content removed from the evaporator. If we assume that some small solids will be present, the standard deviation can be as high as 30% relative. With repeats in the number of samples analyzed, the standard deviation can be expected to fall to 10 - 20% relative.

For accountability, we typically want to know to within 10% (90% confidence), the mass of fissile material moved from tank to tank. For a small number of samples (less than 60), a Student's t statistic must be used, instead of the common "2", as in "2 sigma". Table 1 lists the t statistic at the 90% confidence level versus the number of samples analyzed. The table also shows the percent uncertainty of the mean for two different assumed sample standard deviations.

If the cleaning solution is essentially just a salt solution, the standard deviation can be expected to be near the 5% mark. The third column in Table 1 lists the percent uncertainty in the mean for the different number of samples analyzed. At four samples, the percent uncertainty is under 10% relative to the mean. As shown in the fourth column, when solids are involved in the sample, the number of analyses needed to reduce the uncertainty to under 10% is much higher. At 14 samples, with a sample standard deviation of 20%, the uncertainty in the mean is finally under the 10% mark.

These calculations point to a significant number of samples for the spent cleaning solution. The total number of samples recommended and the dispositions are:

Ship to NRC	4
A&PC analyze immediately	4
Hold for future A&PC analyses	10
Hold for future NRC analyses	10
Hold as archive	7
Total	35

Four of the samples can be shipped immediately to the NRC for analyses. Four other samples shall be immediately analyzed by the WVNS laboratory. If the percent uncertainty of the mean at the 90% confidence level (equals the standard deviation divided by the square root of the number of samples analyzed times a t statistic) is not less than 10% of the average analysis, then additional samples from the remaining 27 shall be analyzed.

Direct any questions about the sampling schedule to the undersigned.

J. L. Mahoney, Senior Engineer, MS-M
IRTS Engineering
West Valley Nuclear Services Co., Inc.

JLM

wp

Table 1
Sample Calculations of the
Percent Uncertainty of the Mean
For Two Different Sample Standard Deviations

Number of Samples	Student's t Statistic	5% Rel. Sample Stdev	20% Rel. Sample Stdev
2	6.314	31.6	126.3
3	2.920	10.3	41.3
4	2.353	6.8	27.2
5	2.132	5.3	21.3
6	2.015	4.5	18.0
7	1.943	4.0	15.9
8	1.895	3.6	14.3
9	1.860	3.3	13.2
10	1.833	3.1	12.2
11	1.812	2.9	11.5
12	1.796	2.7	10.8
13	1.782	2.6	10.3
14	1.771	2.5	9.8
15	1.761	2.4	9.4
16	1.753	2.3	9.1

From: IRTS Engineering
Ext. 4183 MS-M

Letter #: CJ:91:0047

Date: May 28, 1991

Subject: Best Estimate and Uncertainty of Pu in the LWTS Evaporator

To:

P. J. Valenti MS-W

CC:

D. E. Carl	MS-I	P. S. Klanian	MS-I
J. C. Cwynar	MS-W	D. C. Meess	MS-Z26
D. J. Fauth	MS-56	D. K. Ploetz	MS-305
R. F. Gessner	MS-201	C. J. Roberts	MS-D
		MRC (original)	MS-MRC

This memo presents the best estimate of the total grams of plutonium thought to be present in the LWTS evaporator. Based on gross alpha activity data and specific gravity measurements from the evaporator feed and product tanks, approximately 359 ± 100 grams (2 sigma) of plutonium is estimated present in the evaporator.

The conservative assumptions that have guided the SAR update (460 grams of Pu-239/Pu-241) remain valid. During the acid wash phase, the recovered plutonium shall be compared against 359 ± 100 grams. Details of the calculation and the uncertainty handling are provided as attachments to this memo.

Contact the undersigned concerning questions about this estimate.

J. L. Mahoney, Senior Engineer, MS-M
IRTS Engineering
West Valley Nuclear Services Co., Inc.

Attachment A: Estimate of Pu in the LWTS Evaporator

Attachment B: Method of Uncertainty Calculation

Attachment A
Estimate of Pu in the LWTS Evaporator

DISCUSSION

To compute the amount of plutonium thought to be inside the LWTS evaporator, a simple material balance is computed. The outlet Pu mass (or Ci) is compared against the inlet mass (or Ci) with the difference representing accumulation (or loss) of material in the evaporator.

Due to the concentration effect that occurs in the evaporator, the inlet concentrations of mass (gm/ml) or activity (Ci/ml) cannot be compared to the outlet concentrations. The inlet concentrations must be converted to the equivalent outlet conditions (weight % solids and density) before a comparison can be made.

The inlet concentrations restated in terms of the outlet conditions will be:

$$\alpha\ Pu_i\ [\mu Ci/ml] = \frac{(\text{Meas } \alpha\ Pu_i) * (\text{Wt\% outlet}) * (\text{density outlet})}{(\text{Wt\% inlet}) * (\text{density inlet})}$$

The outlet activity concentration is subtracted from the restated inlet Pu activity concentration.

$$\text{delta} = \alpha\ Pu_i\ [\mu Ci/ml] - \alpha\ Pu_o\ [\mu Ci/ml]$$

If the delta is zero, no accumulation is occurring in the evaporator. If the delta is negative, Pu present in the evaporator is being flushed from the evaporator. If the delta is positive, accumulation is occurring inside the evaporator. The total mass or curies that remain in the evaporator is calculated by multiplying the delta times the volume of concentrates that were removed from the evaporator.

To perform the computation with the historical LWTS data, a few more details must be added. To match the drum production, hence the volume of concentrates, averages will be created from the batch runs applicable to each campaign. This allows one loss figure to be quoted for each campaign. Gross alpha measurements will be used to represent alpha Pu since the supernate alpha activity is more than 99% due to plutonium. In this text and the accompanying tables and graph, when alpha Pu measurement is discussed, gross alpha measurement is actually implied.

Also, the weight percent solids in the inlet and outlet streams are not directly measured. Laboratory experimentation has shown that the weight percent solids in the salt solutions are related to the density. With radioactive samples, a measurement of the density provides sufficient information to compute the weight percent solids via:

$$\text{Wt\% TDS} = [\ (\text{density} - 1) / 0.00475\ ]\ \hat{}\ 0.8726$$

The volume of concentrates produced per campaign is computed via the use of 40 gallons of concentrates per cement drum.

A final conversion from curies of alpha-emitting plutonium to grams of plutonium is needed. The conversion, based on the L. E. Rykken topical report which defined the contents of waste tank 8D-2, after aging the material to a nominal mid-range of the STS operations, is 3.68 grams of total Pu per Ci of alpha-emitting Pu.

The equations to compute the grams of Pu that are accumulating in the evaporator during a LWTS campaign are:

$$\alpha\ Pu_i\ [\mu Ci/ml] = \frac{(Meas\ \alpha\ Pu_i) * (calc\ Wt\%\ out) * (density\ out)}{(calc\ Wt\%\ inlet) * (density\ inlet)}$$

$$Calc\ Wt\%\ out = [\ (density\ out - 1)\ /\ 0.00475\ ]\ \hat{}\ 0.8726$$

$$Calc\ Wt\%\ inlet = [\ (density\ inlet - 1)\ /\ 0.00475\ ]\ \hat{}\ 0.8726$$

$$delta = \alpha\ Pu_i\ [\mu Ci/ml] - Meas\ \alpha\ Pu_0\ [\mu Ci/ml]$$

$$\alpha\ Pu\ [Ci] = 3.68 * delta * volume\ of\ concentrates$$

In the above 5 equations, only 4 values are measurements of the process streams (excluding the volume of concentrates). Using standard techniques for combining uncertainties of multiple measurements (see Appendix B), the uncertainty in total grams of Pu deposited in the evaporator during each campaign can be computed.

Table A-1 lists the input activity concentrations and density measurements for the 21 LRTS campaigns along with the volumes of concentrates. The restated inlet activity concentration and other calculated values are shown in table A-2. The total grams of Pu expected in the evaporator is 359 with a two sigma uncertainty of 100 grams.

Table A-1
IRTS Feed and Product Tank Measurements

Campaign Number	15B Density gm/ml	15A Density gm/ml	15B α Pu µCi/ml	15A α Pu µCi/ml	Vol Concent gals
1	1.234	1.321	0.069	0.079	16040
	1.227	1.324	0.024	0.064	
	1.140	1.298	0.044	0.044	
	1.244	1.301		0.066	
2	1.137	1.317	0.026	0.053	15280
	1.174	1.308	0.038	0.067	
	1.242	1.298	0.056	0.054	
	1.245	1.306	0.059	0.079	
3	1.198	1.306	0.042	0.072	22560
	1.204	1.303	0.026	0.067	
	1.239	1.308	0.092	0.069	
4	1.220	1.318	0.044	0.072	13360
6	1.173	1.320	0.058	0.077	13120
	1.145	1.326	0.028	0.054	
	1.094	1.320	0.029	0.038	
7	1.117	1.323	0.022	0.036	23920
	1.071	1.319	0.022	0.051	
	1.078	1.316	0.020	0.037	
	1.068	1.315	0.035	0.050	
8	1.089	1.328	0.028	0.051	27840
	1.064	1.322	0.021	0.052	
	1.082	1.331	0.022	0.051	
	1.062	1.327	0.014	0.055	
	1.085	1.328	0.025	0.061	
9	1.067	1.325	0.018	0.038	27400
	1.079	1.325	0.039	0.067	
	1.080	1.331	0.019	0.059	
	1.057	1.333	0.013	0.051	
10	1.090	1.326	0.029	0.079	5920
	1.038	1.311	0.010	0.059	
11	1.056	1.334	0.013	0.064	25680
	1.077	1.328	0.036	0.056	
	1.062	1.331	0.015	0.053	
	1.085	1.333	0.029	0.056	

Table A-1 (cont.)
IRTS Feed and Product Tank Measurements

Campaign Number	15B Density gm/ml	15A Density gm/ml	15B α Pu µCi/ml	15A α Pu µCi/ml	Vol Concent gals
12	1.081	1.326	0.026	0.061	25720
	1.083	1.321	0.026	0.091	
	1.077	1.309	0.035	0.046	
	1.074	1.324	0.025	0.053	
13	1.088	1.324	0.022	0.064	20000
	1.080	1.321	0.038	0.068	
	1.082	1.314	0.028	0.066	
14	1.081	1.326	0.045	0.068	24440
	1.093	1.327	0.042	0.066	
	1.073	1.315	0.041	0.071	
15	1.091	1.330	0.029	0.064	23680
	1.090	1.327	0.035	0.082	
	1.081	1.333	0.020	0.072	
	1.095	1.328	0.039	0.054	
16	1.064	1.316	0.043	0.095	27360
	1.062	1.317	0.029	0.027	
	1.060	1.328	0.026	0.063	
	1.073	1.324	0.028	0.075	
	1.069	1.334	0.033	0.063	
17	1.066	1.331	0.035	0.064	25720
	1.068	1.328	0.025	0.070	
	1.057	1.329	0.027	0.056	
	1.067	1.333	0.029	0.054	
	1.060	1.333	0.029	0.056	
	1.060	1.326	0.030	0.056	
18	1.064	1.329	0.027	0.046	21960
	1.066	1.331	0.029	0.049	
	1.065	1.313	0.031	0.055	
	1.067	1.314	0.036	0.047	
19	1.053	1.315	0.040	0.056	27000
	1.072	1.306	0.028	0.040	
	1.076	1.316	0.031	0.059	
	1.074	1.319	0.042	0.064	
	1.077	1.307	0.039	0.072	
	1.089	1.302	0.045	0.093	

Table A-1 (cont.)
IRTS Feed and Product Tank Measurements

Campaign Number	15B Density gm/ml	15A Density gm/ml	15B α Pu µCi/ml	15A α Pu µCi/ml	Vol Concent gals
20	1.057	1.299	0.0253	0.0849	26440
	1.071	1.307	0.0522	0.0844	
	1.057	1.324	0.0327	0.0732	
	1.053	1.295	0.0315	0.0599	
	1.065	1.319	0.0381	0.0678	
	1.066		0.0404		
	1.068		0.0596		
	1.069		0.0420		
	1.058		0.0251		
	1.050		0.0231		
	1.056		0.0484		
	1.063		0.0424		
	1.052		0.0310		
21	1.051	1.317	0.0271	0.0503	2280
	1.064		0.0323		

Table A-2
IRTS Feed and Product
Pu Accountability Values

Campgn Number	Restated Inlet α Pu µCi/ml	Outlet α Pu µCi/ml	Delta α Pu µCi/ml	Concentr Vol gals	Delta gm Pu	Uncert gm Pu
1	0.046	0.058	0.012	16040	2.6	1.6
2	0.045	0.063	0.006	15280	1.3	1.3
3	0.053	0.069	0.007	22560	2.1	2.3
4	0.044	0.072	-0.005	13360	-1.0	2.8
6	0.038	0.056	0.036	13120	6.6	1.4
7	0.025	0.044	0.060	23920	20.0	2.2
8	0.022	0.054	0.042	27840	16.4	2.3
9	0.022	0.054	0.050	27400	18.9	2.6
10	0.020	0.069	0.026	5920	2.1	0.8
11	0.023	0.057	0.051	25680	18.1	2.4
12	0.028	0.063	0.054	25720	19.2	2.4
13	0.030	0.066	0.051	20000	14.3	2.2
14	0.042	0.068	0.103	24440	35.1	2.8
15	0.031	0.068	0.047	23680	15.5	2.2
16	0.032	0.065	0.095	27360	36.4	2.4
17	0.029	0.059	0.095	25720	34.2	2.1
18	0.031	0.049	0.104	21960	31.7	2.2
19	0.038	0.064	0.101	27000	38.0	2.2
20	0.038	0.074	0.120	26440	44.2	2.1
21	0.030	0.050	0.[illegible]	2280	3.6	0.4

Attachment B
Method of Uncertainty Calculation

DISCUSSION

Classical uncertainty calculations begin with the recognition that true random normal variation in each variable is required. All the variables under study must also be independent. When these conditions are met, calculation of the uncertainty begins with partial derivatives of the governing equation.

For this memo, the governing equations were depicted at the end of Attachment A. When combined into one global equation, the grams of Pu lost during every batch is:

$$\text{gms Pu} = 3680 * \text{Vol} * \{Pu_I * [(D_0-1)/0.00475]^{0.8726} * D_0 + [(D_I-1)/0.00475]^{0.8726} + D_I - Pu_0\}$$

where Vol: liters of concentrate
Pu_I: measured alpha Pu [μCi/ml] in evaporator inlet
Pu_0: measured alpha Pu [μCi/ml] in evaporator outlet
D_I: measured density [gm/ml] of evaporator inlet
D_0: measured density [gm/ml] of evaporator outlet

Two major types of measurement errors are considered in this analysis: error on measurement of the densities and error in the measurement in the alpha Pu activity concentrations. The error in the correlation tying density to weight % solids is not included in this analysis. The uncertainty due to the four measurements is thought to dominate the total uncertainty, allowing this contribution to be discarded. Also, the volume of concentrates is thought to be known so well that its uncertainty can be excluded in the total uncertainty calculation.

For a generic governing equation, the combined uncertainty is created via partial derivatives. For example if the following equation were used:

$$F = X * Y - Z$$

the uncertainty in F is calculated from

$$\sigma_F^2 = \sigma_X^2 * (\delta F/\delta X)^2 + \sigma_Y^2 * (\delta F/\delta Y)^2 + \sigma_Z^2 * (\delta F/\delta Z)^2$$

where: σ_F = uncertainty in F (likewise in X, Y, & Z)
$\delta F/\delta X$ = partial derivative of F with respect to X (likewise for Y & Z)

In our governing equation four partial derivatives are needed. The fully developed equation is too lengthy to be presented here. The math is left for the interested reader.

The uncertainties in the governing equation should theoretically come from a significant number (>20) of replicate samples & analyses of a typical evaporator batch. All that is available, however, are the density and gross alpha measurements for the individual batch runs in each campaign. The multiple measurements presented in table A-1 can be used estimate the uncertainty in the gross alpha and density measurements.

As operated, the LWTS evaporator produced batches that could arguably be exact replicates from campaign 1 through campaign 21. The argument for similar batches would suggest that the density differences between batches is controlled by the uncertainty in the measurement. Stated differently, this implies that the true batch-to-batch variance is small compared to our ability to measure the density. This means the standard deviation between batches can be used to estimate the actual measurement uncertainty. The same argument will be extended to gross alpha measurements although the variability in the evaporator loss mechanism could effect the standard deviation estimate. Table B-1 and Figure B-2 present the standard deviations computed from this collection of data.

The true measurement uncertainty can be estimated from the multiple standard deviation measurements. A mean-value calculation would incorrectly over-estimate the true standard deviation (since standard deviations are distributed per a chi-square function). An approximation to the true standard deviation is to take the logarithm of the standard deviations, average the log values, and calculate the value of ten raised to that average (data presented in table B-1). The best estimate in the uncertainties for the density and gross alpha measurements are ± 0.006 gm/ml and ± 0.008 µCi/ml respectively.

Once the losses and uncertainty in the losses for all the campaigns are computed, the grand total and its uncertainty must be computed. The total loss of Pu into the evaporator is simply the sum of the individual campaigns. The uncertainty, however, must be computed via the partial derivatives of a governing equation. In this case, the governing equation is simply:

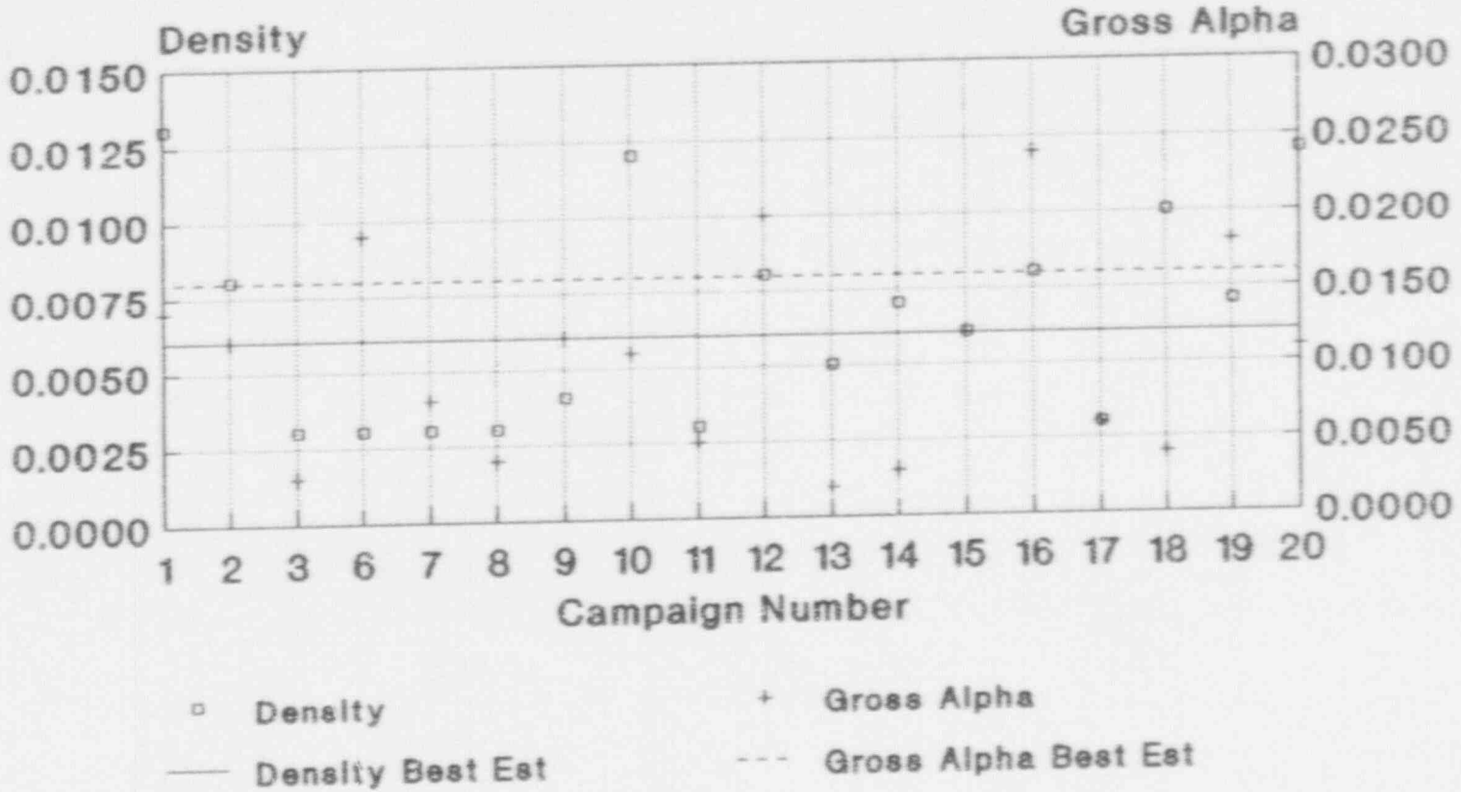
$$\text{Total} = \#1 + \#2 + \#3 + \ldots + \#20 + \#21$$

The partial derivatives are all equal to 1. So the combined total uncertainty is the square root of the sum of the squares of the individual uncertainties.

Table B-1
Best Estimates of Standard Deviations for
Evaporator Concentrates Density and Gross Alpha Measurements

(1) Campaign Number	(2) Density Std Dev	(3) Gross Alpha Std Dev	(4) Log of Col (2)	(5) Log of Col (3)
1	0.013	0.014	-1.873	-1.844
2	0.008	0.012	-2.108	-1.907
3	0.003	0.003	-2.599	-2.593
6	0.003	0.019	-2.460	-1.716
7	0.003	0.008	-2.542	-2.088
8	0.003	0.004	-2.485	-2.362
9	0.004	0.012	-2.385	-1.922
10	0.012	0.011	-1.933	-1.967
11	0.003	0.005	-2.577	-2.319
12	0.008	0.020	-2.118	-1.702
13	0.005	0.002	-2.290	-2.745
14	0.007	0.003	-2.177	-2.590
15	0.006	0.012	-2.188	-1.928
16	0.008	0.024	-2.121	-1.611
17	0.003	0.006	-2.548	-2.196
18	0.010	0.004	-2.019	-2.401
19	0.007	0.018	-2.172	-1.746
20	0.012	0.011	-1.903	-1.967
average			-2.250	-2.089
10^{average}			0.006	0.008

Best Estimate of Standard Deviations
Evaporator Concentrates
Density & Gross Alpha



Density Best Est: 0.006 gm/ml
Gross Alpha Best Est: 0.008 uCi/ml

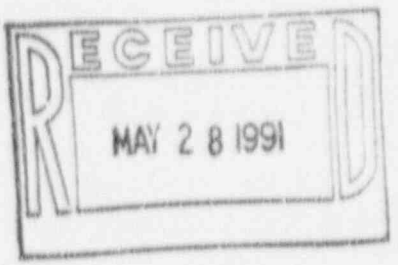
JLM 5/24/91

DC:91:0055 ATTACHMENT B May 29, 1991

FA:91:0049
RS:91:0024

bcc:

J. J. Buggy	MS-07
J. C. Cwynar	MS-W
E. R. Fallace	MS-D
R. A. Humphrey	MS-42
R. J. Johnson	MS-D
P. S. Klanian	MS-I
J. L. Knabenschuh	MS-41A
D. H. Kurasch	MS-307
W. G. Poulson	MS-307
J. J. Prowse	MS-D
C. J. Roberts	MS-D
D. L. Smithmeyer	MS-52
S. J. Szalinski	MS-42
P. J. Valenti	MS-W
Job File	MS-D
FA Letter Log	MS-41A
RS Letter Log	MS-42



RLW4300:SHA-163

West Valley
Nuclear Services Company
Incorporated

ACTION:
WD:2087

WD:91:0555
P.O. Box 191
West Valley, New York 14171-0191

May 24, 1991

MS-D

Mr. T. J. Rowland, Director
West Valley Project Office
U.S. Department of Energy
MS-DOE
P.O. Box 191
West Valley, New York 14171-0191

Dear Mr. Rowland:

SUBJECT: Request for Approval of LWTS SAR-005, Rev. 2, Draft C

References: 1) Letter WD:91:0487, J. L. Knabenschuh to T. J. Rowland, "Request for Approval of the Revised LWTS (SAR-005)," dated May 3, 1991

2) Letter CBL:018:90-0785:91:10 (DW:91:0341), T. J. Rowland to J. J. Buggy, Jr., "West Valley Project (WVPO) Comments on Revised LWTS (SAR-005)," dated May 15, 1991

The purpose of this letter is to request DOE approval of the attached Safety Analysis Report for the Liquid Waste Treatment System (WVNS SAR-005, Rev. 2, Draft C) which addresses safety and health protection matters regarding the acid cleaning of the LWTS evaporator. This revision incorporates responses to your comments on Draft B which were summarized in Reference 2. Only the pages which have been revised since Rev. 1 are included as Attachment A.

Attachment B summarizes the resolution of each of the fourteen items of Reference 2. Attachment C is a copy of Reference 2 showing the identification by number of each of the fourteen comments. These identification numbers appear in the margin of Attachment A to indicate where the most recent changes have been made (revising Draft B to Draft C.)

DOE approval is requested by May 31 so that the acid cleaning may proceed on schedule. Upon receipt of DOE approval, WVNS will release Revision 2 of the subject SAR.

RLW4300.SEA-163

If you have any questions, please contact Joe Johnson at Extension 4064.

Very truly yours,

John L. Kr[illegible]benschuh
Vice President and Manager
Environmental Safety, Health and Quality Assurance
West Valley Nuclear Services Co., Inc.

FA:91:0049
RS:91:0024

RJJ:rlw

Attachments: A) WVNS-SAR-005, Rev. 2 (revised pages only)
B) Resolution of Items from Reference 2
C) Letter CBL:018:91-0785:91:10 (with items numbered)

cc: E. A. Matthews, DOE-WV, MS-DOE
R. B. Provencher, DOE-WV, MS-DOE
J. A. Yeazel, DOE-WV, MS-DOE

RLW4300:SEA-163

ATTACHMENT A

TABLE OF CONTENTS (CONTINUED)

VOLUME IV

LIST OF TABLES

H.2.3 RADIOLOGICAL IMPACT FROM ABNORMAL OPERATIONS

Abnormal operations are events which could occur from malfunctions of systems, operating conditions, or operator error. Abnormal events are only of consequence for those systems in the LWTS which process, control, or confine radioactivity. The abnormal events considered (Section H.9.1) are of little consequence in terms of potential for environmental releases.

H.2.4 ACCIDENTS

Seven accidents associated with the operation of the LWTS were analyzed (Section 9.6). Accidents analyzed include ruptures of various process lines, a tank rupture inside and outside XC3, HEPA filter failure, HEPA filter fire for the Vessel Off-Gas (VOG) System and the Main Plant Ventilation System, and a spill of evaporator cleaning solution following a pipe rupture. Releases are assumed to be direct to atmosphere for the various pipe ruptures, while all other releases are from the Main Plant Stack. Doses from these accidents to the maximally exposed off-site individual for the two process line ruptures are 21 and 0.86 mrem, respectively. The difference between the two accidents concerns the type of material being handled and quantity of material released. For the tank rupture outside XC3 (5D15A1 in UPC), the dose is calculated as 2.1 E-03 mrem. Analysis of a tank rupture accident inside XC3 (Spent Zeolite Tank, 71-D-007) leads to a predicted dose of 1.3 E-05 mrem to the maximally exposed off-site individual. The HEPA failure and fire accidents are assumed to be conservative since the 8D-2 Tank activity is the starting point for the analysis instead of decontaminated supernatant. The dose to the maximally exposed off-site individual for the VOG System HEPA failure is 0.34 mrem while the HEPA fire dose is projected to be an order of magnitude higher at 3.4 mrem. The dose for a failure of or fire in the Main Plant Ventilation HEPAs is considered to be the same as or less than that for the VOG HEPAs. The dose from a spill of evaporator cleaning solution following the rupture of a transfer pipe is calculated as 120 mrem to the maximally exposed off-site individual and 2.2 rem to the maximally exposed on-site individual.

- o Spent evaporator cleaning solutions

H.4.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA FOR THE LWTS

The majority of LWTS components will be installed in existing structures such as XC3, PPC, ULO, and UPC. All existing structures have been analyzed for structural and mechanical safety in previous documents (NFS, 1962 and WVDP, 1985a). Minor modifications of structural materials necessary to accommodate LWTS piping and equipment in the Process Building are not considered significant enough to change the overall structural safety in the above mentioned analysis. Furthermore, the hazards associated with a loss of structural integrity are minor when LWTS is compared to previous reprocessing activities. The only structure which can be considered as new is the concrete below-ground trench which serves as a pipe chase between Tank 35104 and LWTS. Seismic analysis of the trench was not considered necessary due to the relatively small quantity of activity present at any one time, and the low off-site dose potential when an accidental spill was considered (Section H.9.6.1.5).

H.4.3 SAFETY PROTECTION SYSTEMS

H.4.3.1 GENERAL

LWTS has been designed to meet all applicable engineering codes and standards throughout construction, installation, and system operation (WVDP, 1985b). Specific information regarding the various types of safety protection systems considered necessary for the operation of LWTS appear below.

H.4.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

The primary confinement barrier for the relatively low concentration of radioactive materials processed by LWTS operations is considered to be the system piping, tankage, and associated process vessels. Secondary confinement is afforded by Extraction Cell 3 (XC3) which houses the heart of the LWTS system, the high efficiency waste evaporator, and the ion exchange columns.

H.4.3.5 RADIOLOGICAL PROTECTION

Construction and maintenance activities will be performed in accordance with WVDP-10, "Radiological Controls Manual." Shield walls, confinement and containment structures as well as administrative controls (procedures, training, OSRs, etc.) will be used as necessary to maintain radiation doses to occupationally exposed personnel ALARA. Protective clothing (anti-C's, respiratory protection) will be worn when required by radiological conditions. In addition, precautions such as system decontamination and flushing are required in the case of necessary contact maintenance.

H.4.3.6 FIRE AND EXPLOSION PROTECTION

The LWTS will have fire detection equipment, alarm systems, and suppression systems commensurate with potential fire hazards associated with LWTS operation as determined by the WVNS Radiation and Safety group.

H.4.3.7 RADIOACTIVE WASTE HANDLING AND STORAGE

LWTS operations will use a number of existing tanks to temporarily store LWTS feed material and process material in different stages of the operation. Storage of these liquid materials in any one process vessel or tank is considered to be a temporary condition requiring no additional safety analysis.

H.4.3.8 INDUSTRIAL AND CHEMICAL SAFETY

The WVNS Industrial Hygiene and Safety Manual, WVDP-011, will provide the administrative guidelines and control over industrial and chemical safety in regard to the operation of LWTS. The LWTS process will provide for addition of acid or caustic to feed solutions. Analysis of an accident where acid and caustic are inadvertently mixed is considered in Section H.9.7. Analysis of an accident where an acid cleaning solution is spilled is considered in section H.9.6.11.

H.4.4 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Safety and service classifications of important structures, components, and systems associated with the LWTS are presented in Table H.4.4-1. The procedures and criteria used to arrive at the designated classifications are contained in Section H.4.4 of Volume I.

Accident analysis (Section H.9.6) shows that no postulated accident will result in a dose to the maximally exposed off-site individual approaching 500 mrem. Therefore, no items need to be classified as Safety Class A or B. The heating/ventilation (HV) supports, vessel off-gas piping, and ventilation instrumentation and alarms are classified as Safety Class C. The HV supports and VOG piping are Class C because they ensure confinement of radioactivity. The instrumentation and alarms for ventilation systems are considered Class C because their loss of function could allow unmonitored releases. Likewise, airborne particulate and area radiation monitors are Class C due to the potential for unmonitored releases should failure occur. 1

Sump piping, instrumentation, and alarms are Class C for reasons of confinement of radioactive material and because loss of instrumentation and alarms could lead to undetected leakage and potential for inadvertent exposure. All other items are Safety Classified as N.

H.4.5 DECONTAMINATION AND DECOMMISSIONING

Specifications for LWTS equipment and components which will contain radioactive material call for stainless steel (types 304, 304L, 316, 316L). This will allow periodic flushing of the system with strong decontamination agents, thus removing the maximum amount of deposited radioactive material. As a result, personnel dose will be minimized during the decommissioning effort. It must be recognized that the LWTS system would most likely be the last of the WVDP systems to be dismantled due to the key role it plays in waste water management.

pumped to the zeolite ion exchanger (71-D-003) via Pump 71-P-15. When LWTS construction has been completed, the ion exchanger effluent will be sent to one of two 34,000 litre monitoring tanks. After sampling, the liquid will be routed to either Lagoon No. 4 or 5 or the Interceptors depending upon radioactivity concentrations (see Section H.6.1.1.8). Prior to completion of this system effluent from 71-D-003 will be recycled or fed to the existing LLWTF (02 Plant) for further processing and release to the environment.

Evaporator distillate is pumped from the distillate surge tank via Pump P-15 to a zeolite ion exchanger which is solely devoted to evaporator distillate processing. This ion exchanger is equipped with a differential pressure monitor and effluent radiation monitor similar to those discussed in Section H.6.1.1.6 for the filter/ion exchange train. Off-spec effluent (>3E-07 μCi/mL gross beta) will also be diverted to Tank 5D15B or the Interceptors via the PPC manifold.

After extended periods of high TDS evaporator operation it may be necessary to perform a cleaning operation to remove accumulated solids. The solids increase radiation background near the evaporator, lower its boiling capacity and accumulate fissile isotopes in the evaporator scale or sludge in the bottom head. The solids form because of chemical changes to dissolved salts as they are heated. Since these solids will not dissolve in water alone it is necessary to use up to 2M nitric acid (≤12%) to convert them into a soluble form. The analyses in section 9.8.1 indicate that criticality is not a concern during evaporator cleaning. However, approximately one gram per liter of boron (as boric acid) may be added to act as a neutron poison for fissile isotopes which may be present.

The cleaning solution is placed in the evaporator and heated until sampling results reflect limited effectiveness of further scale dissolution. Condensate is returned to the evaporator to maintain a constant liquid level. The spent solution containing the dissolved solids is cooled and transferred to a holding tank for sampling. This solution will not be suitable for cementing in CSS because of high activity and cement recipe

qualification requirements; it will be routed back to the high level waste tank (8D-2) after pH adjustment. All temporary holding tanks and transfer lines will be flushed and/or sparged.

Airborne concentrations of NO_x and nitric acid fumes at the main plant stack release point will be less than 36 ppm and 5 ppb, respectively, over a four hour period (Burn, 1991). These concentrations would be approximately 66% higher if the VOG scrubber and condenser were to become inoperable. Atmospheric dispersion will further reduce the concentrations (well below the applicable threshold limit values-time weighted average) at potential on-site and off-site receptor locations. No impact on the integrity of the HEPA filters is expected at such low concentrations of nitric acid fumes. 5

The neutralized solution will be returned to Tank 8D-2. Therefore, this process will not result in any additional liquid releases to the environment. The cleaning operation will remove radionuclides held up from other waste streams (mostly decontaminated supernatant and sludge washes). Therefore, no net increase in airborne radionuclide emissions is anticipated from this operation.

Depending on the accumulation of solids, cleanouts may be performed more than once over the life of the system.

H.6.1.1.3 Concentrates Collection and Transfer

Evaporator concentrates leave the evaporator at approximately 109°C. They are cooled to approximately 35°C in the concentrates cooler and pumped to Tank 5D15A1 or 5D15A2 for temporary storage. Process control limits for this waste stream are as follows:

- High TDS liquids (in the range of 500 ppm to about 40 w/o depending upon the ionic species) will be concentrated to a maximum of 46 w/o dissolved solids.

REFERENCES FOR SECTION H.6.0

American National Standards Institute (ANSI) N13.1-1969 (R1982), Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity In Effluents.

Burn, P., 1991, Memo EO:91:0055 to G. G. Baker, "NYSDEC Stack Release Permit Modification for LWTS Evaporator Cleaning," dated April 26, 1991.

Saha, A. K., 1986, Document WVNS-PN-003 to Distribution, "LWTS Process Control Interlocks," Rev. 0, dated March 19, 1986.

WVDP, 1985, Safety Analysis Report, Volume II, Existing Plant and Operations.

WVDP, 1985a, Safety Analysis Report, Volume IV, Cement Solidification System.

WVDP, 1985b, Safety Analysis Report, Volume III, Vitrification System.

H.9.4 ABNORMAL EVENTS - OFF-GAS TREATMENT

Failures associated with the Vessel Off-Gas system could cause greater than normal releases of radioactive material. These failures include development of a leak in the HEPA filter or reduced capacity due to high loading or excessive moisture. These conditions are detected by differential pressure instrumentation across each filter. Additional HEPA filtration downstream of the VOG HEPAs exists to prevent any activity which leaks by the VOG HEPA to be filtered prior to discharge up the main plant stack. Another abnormal event would be the failure of one of the VOG blowers (6K-2 or 2A). Since a duplicate blower exists, the system can be operational at all times and provide for uninterrupted service during filter changeout. Additional abnormal events for this system have been described in Table B.9.1-2 of Volume II.

H.9.5 ABNORMAL EVENTS - VENTILATION SYSTEM

Failures associated with the main plant ventilation are similar to those of the VOG system. A HEPA filter develops a leak, or roughing and HEPA filters develop high loadings. Other failures include damper failure or Heating/Ventilation blower failure. Problems with HEPA filtration increase the potential for increased release of activity. Instrumentation exists to detect filter failure or high differential pressure. Damper failure or blower failure will cause loss of ventilation and possible air reversals. Like the VOG, two ventilation trains exist for the Main Plant Ventilation to ensure uninterrupted service.

H.9.6 ACCIDENTS

H.9.6.1 ACCIDENTS ANALYZED

Seven major accidents have been analyzed in this section. They include: 1) Rupture of a process line between the process building and the 01-14 building; 2) Rupture of a process line in the LWTS pipe chase and/or the pipe chase leaving the 01-14 Building; 3) Two tank rupture scenarios involving a tank inside XC3 and one outside XC3; 4) Vessel Off-Gas HEPA filter failure; 5) Vessel Off-Gas HEPA filter fire; 6) Main ventilation HEPA filter failure or fire; and 7) Spill of evaporator cleaning solution following a pipe rupture.

The above listed accidents are considered to represent the most severe case for procedural or equipment failure in the LWTS. LWTS has been designed to preclude such occurrences; nowever, the probability of occurrence is still considered greater than zero for the above cases.

H.9.6.2 SOURCE TERMS

The source terms for the above spill-related accidents are highly dependant upon where in the process sequence the event occurs. Evaporator cleaning solution will have a relatively higher concentration of uranium and plutonium (up to 120 μCi/mL) than other waste streams. The second largest source term for accidents involving LWTS systems is considered to be the evaporator concentrates, which can reach 50 μCi/mL of Cs-137 under current design specifications. Another large source is the concentrator feed make-up tank overheads fed to LWTS at a concentration of 19 μCi/mL of Cs-137. Waste streams other than these sources will be reduced by various decontamination factors associated with the LWTS processing equipment to levels significantly below 19 μCi/mL . For the HEPA filter failure and the filter fire, the nuclide distribution is assumed to be represented by those nuclides found in the 8D-2 supernatant. The nuclides and activities released for each accident are given in Table H.9.6-1 through H.9.6-7.

H.9.6.3 OFF-SITE RADIATION DOSES

All seven accidents analyzed will result in the off-site release of radioactivity. In the case of spills, the material released would primarily be in aerosol or vapor form resulting from the evaporation of the aqueous phase. A HEPA filter fire would result in an off-site release as would the loss of the VOG HEPA filter integrity due to high dust loading or wetness. In the case of fire, the integrity of the downstream filter is assumed to remain intact with a diminished DF of 100. For filter failure, the downstream filter is not

The dose to the maximally exposed off-site individual is given in Table H.9.6-1. A spill of 4500 L of evaporator bottoms and a release of 8.3E-02 curies to the environment is calculated to contribute 21 mrem over the two hour exposure period. The dose conversion factors (DCFs) used for these calculations are taken from WVDP (1987).

H.9.6.5 RUPTURE OF THE LWTS PIPE CHASE AND/OR THE PIPE CHASE LEAVING THE 01-14 BUILDING

As discussed in Section H.9.6.4, rupture of these lines can be caused by a variety of external sources. At the time of this pipe rupture, Tank 35104 is assumed to be at capacity and the evaporator feed pump (71-P-01) is actively pumping decontaminated supernatant to the evaporator at the rate of approximately 200 litres/min. The pump is again assumed to be operating unnoticed for one hour following the rupture spilling 11,000 L of decontaminated supernatant into the STS trench. The trench is conservatively assumed to be open to the atmosphere at the time of the spill. The decontaminated supernatant is assumed to be released at a temperature of 27°C with evaporation taking place at an average temperature of 20°C. The waste stream is assumed to have a total activity of 10.9 μCi/mL with the Cs-137 concentration at 3.21 μCi/mL .

As in section H.9.6.4, the probability of such a rupture is unlikely. Emergency procedures exist which allow for systematic shutdown in the event of an accident. In addition, the process line itself is routed through a sealed concrete trench. Therefore, releases to the atmosphere are severely restricted by the enclosed trench when compared to free atmosphere, as was assumed in the off-site dose calculation. The dose to the maximally exposed individual is 0.86 mrem (Table H.9.6-2). A calculation was also performed considering the CFMUT overheads as the source. The off-site dose from this source was calculated at 0.32 mrem (Dooley, 1986).

See also section H.9.6.11 for the analysis of a spill of evaporator cleaning solution.

127 ft. elevation and is positioned over many of the lines feeding the evaporator. As stated above, damage or disruption of these lines is assumed to have little or no additional impact since most system valves can be activated automatically or manually in the case of an emergency.

The zeolite ion exchanger is located at the 136 ft. elevation of XC-3. A portion of this tank is directly over the Spent Zeolite Tank 71-D-007. At the time of failure, the zeolite ion exchange tank and the spent zeolite tank are assumed to be at maximum capacity of 4,000 litres each, and to contain 4 curies of Cs-137 each, based on the maximum anticipated resin loading of 1 μCi/mL. The zeolite ion exchange tank is assumed to fall down, striking the spent zeolite tank and causing both tanks to fall to the floor of XC-3 at elevation 100 ft. The entire contents of 8,000 litres are assumed to spill onto the floor. The dose to the maximally exposed off-site individual is estimated to be 4.3 E-06 mrem as a result.

The total liquid volume which could be accommodated on the XC-3 floor (assuming the sump is inoperable) before liquid would overflow into the CAA is approximately 18,000 litres. Therefore more than twice the volume of liquid associated with the maximum spillage resulting from vessel support failure can be accommodated and contained within XC-3.

H.9.6.11 RUPTURE OF LWTS TRANSFER PIPE AND SPILL OF EVAPORATOR CLEANING SOLUTION

7

As described in Section H.6.1.1.10, the evaporator located inside XC3 may require periodic cleaning to remove accumulated scale. This operation will be conducted by recycling approximately 3600 L of borated nitric acid solution (up to 2 M) inside the evaporator at 100 C. No significant concentration of this solution will take place. The acidic liquid will then be cooled in the concentrates cooler, and diluted with two evaporator flush volumes (3600 L each) for a total volume of approximately 11,000 L. After transfer to the neutralization tank, the pH will be adjusted such that reprecipitation will not occur. The resulting solution will be sampled and returned to Tank 8D-2.

12

The probability of a spill of this solution is very low due to the short duration of the operation relative to other LWTS processes. However, the spent solution is expected to contain a higher amount of actinides than normal LWTS waste streams (Cwynar, 1991). Prior to the first cleaning of the evaporator the mass of uranium accumulated in the scale was estimated to be less than 110 kg (1.7% U-235). The total mass of plutonium is estimated to be less than 460 g. As much as 40 Ci of mixed fission products may also be present in the scale. To be conservative, this activity is assumed to be entirely from Sr-90. Following dilution with two flush volumes, the radionuclide concentration in the cleaning solution is expected to be less than 120 μCi/mL.

The accident scenario is similar to the one described in Section H.9.6.5. In this case, however, pump 71-P-06 is assumed to be pumping the dilute cleaning solution from Tank 5D-15A2 to Tank 35104 when the pipe rupture occurs. The entire 11,000 L are assumed to spill to the transfer trench. Seventy-three liters of liquid are conservatively assumed to evaporate during a two-hour release period. A partition coefficient of 1000 between liquid and vapor phases was applied to the activity in the evaporated liquid.

The dose to the maximally exposed off-site individual, using dose factors from WVDP-065 (Rev. 2), was extimated to be 120 mrem (Table H.9.6-7). The dose to the maximally exposed on-site individual (assumed to be 100 m from the spill) was estimated to be 2.2 rem.

Non-radiological releases from this accident (nitric acid and NO_x) would result in concentrations below the applicable TLV-TWA at the site boundary.

H.9.7 MINOR ACCIDENTS

A number of minor accidents such as valve failures, pump failures, loss of control panel signals for various operating conditions and mixing of acid and caustic were analyzed. No single failure or series of multiple failures produced results comparable to the major accidents above in terms of dose to the maximally exposed off-site individual. Inadvertent mixing of acid (HNO_3) and caustic (NaOH) in significant quantity could create an explosive mixture

capable of producing severe injury and possibly death. Administrative procedures will adequately protect against such an occurrence. Also, the acid and caustic addition to Tanks 7D-2, 5D15B, 5D15A1 and 5D15A2 is such that both positive displacement pumps cannot be running simultaneously due to a manual interlock.

H.9.8 POTENTIAL FOR NUCLEAR CRITICALITY

Decontaminated supernatant fed from STS will be evaporated and thus concentrated by LWTS processing. The resulting concentrations of U-233, U-235, Pu-239 and Pu-241 must be determined based on the design limitations of the evaporator's ability to concentrate decontaminated supernatant. The concentrations of fissile radionuclides at the processing limit of 46 w/o solids is shown in Table H.9.8-1.

Table H.9.8-2 shows the maximum allowable concentration of fissile material that is administratively allowed in a process vessel. To ensure these limits are not exceeded within the confines of the Liquid Waste Treatment System, batch samples will be collected and analyzed for fissile content at various points in the process. The total maximum concentration of all fissile material in evaporator concentrates is not expected to be greater than 6.6 E-04 grams per litre (O'Ahoofe, 1985). This is well below the most limiting case from Table H.9.8-2, where the uranium concentration is 0.05 g/L and the plutonium concentration is 1.63 g/L, for a total concentration of fissile isotopes of 1.68 g/L. All other combinations of fissile uranium and plutonium nuclides have subcritical concentration limits greater than this value.

The accumulation of fissile materials in evaporator scale is discussed in the next section.

H.9.8.1 EVAPORATOR ACID WASH

Based upon trend analysis of process sample data (Cwynar 1991) following supernatant treatment, it was conservatively estimated that not more than 460 grams total of Pu 239 plus Pu-241, 1.88 kg of U-235 and 105.9 kg of U-238 could be present in the evaporator. To ensure that no potential for criticality exists, bou[illegible] criticality calculations were performed by Yuan (1991) using the SCALE/KENO V computer code system and confirmed by Caldwell (1991) using the MCNP computer code. A right circular cylinder (radius 60 cm) with sides and bottom of 1/2 inch stainless steel (i.e., stainless steel can) was used to approximate the reboiler portion of the evaporator for the mathematical simulation.

It was further assumed that all the fissile material was dissolved in water in a hemisphere on the bottom surrounded by water. Yuan performed a search for the highest k_{eff} assuming a fixed mass (460 grams of Pu-239) and only varied the radius of the hemisphere which varied the density of the Pu for a given total mass. Caldwell further studied the sensitivity of the k_{eff} calculations by varying the amount of fissile plutonium by $\pm$ 10%. These results established the stability or relative sensitivity of the k_{eff} to changes in the fissile mass. Additionally Caldwell examined the effects on k_{eff} if the stainless steel can was reflected by water below the hemisphere in the can. These calculations provide assurance of subcriticality even under the assumption of the evaporator cell flooding. Similar calculations were performed using both Pu and U.

Results of these calculations show that the $k_{eff} + 2\sigma$ of the evaporator containing the fissile material remains less than 0.95.

Realistic considerations of mathematically characterizing the results of acid washing/cleaning the evaporator would include estimates of additional salts being solubilized as well as a much more dilute solution resulting in further reductions of k_{eff}. Both Yuan and Caldwell performed calculations considering the total amounts of fissile material dissolved in water in the bottom of the evaporator (reduced right circular cylinder of radius 60 cm and associated height to optimize density and neutron leakage). The results confirmed the negative effects (lowering the k_{eff}) of the dilution and changing from the hemispherical geometry to the cylindrical geometry. Parks and Dyer (1991) provided an independent validation of the analytical approach. 10 11

The conservative mathematical assumptions used for the bounding criticality calculations envelope the maintenance activity of acid washing/cleaning the evaporator.

REFERENCES FOR SECTION H.9.0

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West Valley Demonstration Project, 1985a, Volume II, Project Safety Analysis Report, Existing Plant and Operations.

West Valley Demonstration Project, 1987, Radiological Parameters for Assessment of West Valley Demonstration Project Activities," WVDP-065.

Yuan, Y. C., 1985, Letter HE:85:0156, Y. C. Yuan to C. J. Robert, dated July 31, 1985, "Evaporation from a Liquid Spill."

Yuan, Y., 1991, Memo FB:91:0081 to C. J. Roberts, "Criticality Evaluation IRTS Evaporator Draft, Final Report," dated April 11, 1991.

TABLE H.9.6-7

DOSE TO THE MAXIMALLY EXPOSED OFF-SITE INDIVIDUAL FOR THE SIGNIFICANT RADIONUCLIDES EVAPORATED FROM A SPILL OF EVAPORATOR CLEANING SOLUTION FOLLOWING RUPTURE OF A TRANSFER PIPE

Nuclide	Activity in Spill (Ci)[2]	Partition Coefficient[3]	Source Term (Ci)[4]	C.E.D.E. (mrem)[5]
Sr-90	4.0 E+01	1000	2.7 E-04	8.3 E-02
Pu-238	9.3 E+01	1000	6.3 E-04	6.8 E+01
Pu-239	2.2 E+01	1000	1.5 E-04	1.8 E+01
Pu-240	1.7 E+01	1000	1.1 E-04	1.3 E+01
Pu-241	1.1 E+03	1000	7.7 E-03	1.8 E+01
TOTAL	1.3 E+03		8.8 E-03	1.2 E+02

1 The significant radionuclides are identified by the product of the source term (Ci) and dose conversion factors (rem/Ci) for ground-level releases. Those radionuclides which contribute more than 0.1% of the total dose are included.

2 Per discussion in text, 11,000 L of dilute evaporator cleaning solution are spilled to the transfer trench.

3 The partition coefficient is the ratio of the activity concentration in the liquid to that in the vapor above the liquid pool.

4 The volume of liquid that evaporates over a two hour release period is estimated to be 73 L.

5 NOTE: Dose factors are from WVDP-065, Rev. 2 (1990). The CEDE to the maximally exposed on-site individual will be a factor of 18 higher.

H.10.4 NORMAL LWTS OPERATIONS

H.10.4.1 LWTS PROCEDURES

LWTS will be operated using procedures prepared, reviewed, and approved per the requirements of the WVNS Policy and Procedures Manual. Procedures will be written to cover all aspects of LWTS operation including but not limited to:

Tank 35104 and Pump P-01
Evaporator 31017
  Distillate Surge Tank
  Concentrate Cooler
  Distillate Pump
  Cleaning
Zeolite Ion-Exchange Operation (003)
Filter
  Low TDS Feed Tank and Pump
  Filter Backwash Tank
Organic and Zeolite Ion-Exchange Vessels (001, 002)
LWC Tanks and Sample Tank (009)
Resin Hopper and Loading Media into 001, 002, and 003 Ion-Exchangers
Unloading Media from 001, 002, and 003 Ion Exchange Vessels
Unloading Solids from Storage Tanks 006, 007, and 008
Concentrate Hold Tanks 5D15A1 and 5D15A2 and Pumps P-05 and P-06
Lagoon 2 and Interceptors
Lagoon 3 and Recycle Water System
Chemical Addition Tanks 14D7 and 14D18
Atmospheric Release Evaporator

FA:91:0049
RS:91:0024

ATTACHMENT B

RESOLUTION OF ITEMS FROM REFERENCE 2 (ATTACHMENT C)

Item No.	Resolution
1	Page 10; H.9.2 has been changed to H.9.6.
2	Page 29; Now reads: "Since these solids will not dissolve in water alone it is necessary to use up to 2M nitric acid (≤12%) to convert them into a soluble form."
3	Page 29; Now reads: "However, approximately one gram per liter of boron (as boric acid) may be added to act as a neutron poison for fissile isotopes which may be present."
4	Page 29; Now reads: "The solids increase radiation background near the evaporator, lower its boiling capacity and accumulate fissile isotope in the evaporator scale or sludge in the bottom head."
5	Page 30; The HEPA's are acid resistant and if the concentration was 66% higher, then the concentration of 8.3 ppb. (1.66 x 5) would still be negligible. Past experience with the VOG system with much higher concentrations for longer periods of time (3 to 6 months) has produced no discernable HEPA filter degradation.
6	Page 30; The trigger levels for future evaporator clean out will be addressed in the next revision of the LWTS SAR prior to resumption of sludge wash processing.
7	Page 69; A new bounding accident has been analyzed and is discussed in Section H.9.6.11, "Rupture of LWTS Transfer Pipe and Spill of Evaporator Cleaning Solution."
8	Page 70; As stated in Section H.9.6.9, "VOG HEPA Filter Fire" the "filter material is fire resistant." The accident as analyzed results in an estimated dose to the maximally exposed off-site individual of 3.4 mrem. Past experience based upon filter changes shows that no significant source of other material (lint) which could further support combination accumulates on the filter.
9	Page 77. This section (H.9.6.11) has been replaced by a new accident analysis and thus the comment is no longer relevant.
10	Page 80: Additional criticality configurations were considered and are discussed on the next page of this revision (p. 81).
11	Page 81; The criticality calculations were conservatively modeled. Following dissolution of the material in the evaporator, subsequent leakage of the dilute solution would only further reduce the k_{eff}.

FA:91:0049
RS:91:0024

12 Page 81. As now stated in Section H.9.6.11 "After transfer to the neutralization tank the pH will be adjusted such that reprecipitation will not occur. The resulting solution will be sampled and returned to Tank 8D-2."

13 Table H.4.1-2 footnote has been removed. There are no changes on this table for this revision (page has been removed).

14 Page 26, H.6.1.1 Changes have been removed. There are no changes in this section for this revision (page has been removed).

ATTACHMENT C

Rec'd.
Rec. Mgmt.
May 15, 1991

Department of Energy
Idaho Operations Office
West Valley Project Office
P.O. Box 191
West Valley, NY 14171
May 15, 1991

DW:91:0341

Mr. J. J. Buggy Jr., President
and General Manager
West Valley Nuclear Services Co., Inc.
P. O. Box 191
West Valley, New York 14171

ATTENTION: J. L. Knabenschuh, Vice President & Environmental Safety, Health & Quality Assurance Manager

SUBJECT: West Valley Project Office (WVPO) Comments on Revised LWTS (SAR-005)

Dear Sir:

Listed below are the WVPO comments on the subject SAR. These comments were discussed with members of your staff and are being transmitted to you as a matter of record.

Item No.

1 o Pg 10, H.4.4, second paragraph; Reference to accident analysis in H.9.2 - looks like accident analysis is in H.9.6.

2 o Pg 29; What is the corresponding Molarity for 12% Nitric Acid?

3 o Pg 29; The sentence describing that the addition of Boron would be prudent doesn't commit to anything. Will Boron be added? If so, how much is a little?

4 o Pg 29, first revised paragraph, second sentence; ...accumulate fissile isotopes in the evaporator scale or sludge in the bottom head.

5 o Pg 30, second revised paragraph, last sentence; Are the HEPA's acid resistant? Will they be unaffected if the concentrations are 66% higher?

6 o Pg 30; The statement is made that future "cleanouts" may be necessary. What is the trigger level for performing another system flush?

7 o Pg 69, H.9.6.1; Accident (7) is an acid wash spill. What about an acid "diversion: to another process piping system? Say through a failed valve alignment? Further, what does the accident analysis selection process look like? How is it decided that the spill is the only credible accident associated with the cleaning process? How about an evaporator overfill? How about acid diversion to a zeolite bed?

Item No.

8 o Pg 70, H.9.6.3; Will a filter fire be enhanced with HNO_3 added to the off-gas stream?

9 o Pg 77, H.9.6.11; "No concentration..." is too limiting. "No significant concentration...or "Concentration should be limited to..." would be preferred.

10 o Pg 80, criticality configurations; Should mention be made of all of the configurations considered? A listing? With a notation that none were more conservations?

11 o Pg 81; Is it necessary to consider criticality if the solutions leaked out of the evaporator to a sump?

12 o Pg 81; Will reprecipitation of fissile material following neutralization cause the fissile material to get "hung up" in other parts of the system on the way back to 8D-2? Has this been evaluated?

13 o TABLE H.4-1-2 - Footnote (1); What does this mean "...not included...to avoid double counting." Where are they included?

14 o H.6.1.1; LWTS is divided into 9 processes... Why is this revised unless something below was added?

Please contact me on extension 4780 if any of these comments require additional clarification.

Sincerely,

T. J. Rowland, Director
West Valley Project Office

cc: J. E. Solecki, DOE-ID
D. B. Engelman, DOE-ID

CEL:018:91 - 0785:91:10

CEL/sl

WD:91:0596

West Valley
Nuclear Services Company
Incorporated

P.O. Box 191
West Valley, New York 14171-0191

RESPONSE:
DW:4410

May 31, 1991

Mr. T. J. Rowland, Director
West Valley Project Office
U.S. Department of Energy
MS-DOE
P.O. Box 191
West Valley, New York 14171-0191

Dear Mr Rowland:

Attention : J. A. Yeazel

SUBJECT : Response to NRC Comments on Cement Recipe

REFERENCE : 1) DOE letter number WSK:015:91, T. J. Rowland (WVPO) to J. J. Buggy (WVNS), "NRC Response to the West Valley Cement Recipe Data Transmittal", dated May 7, 1991

The purpose of this letter is to provide WVNS's response to the NRC comments (transmitted by WVPO in reference 1) regarding the "January 1991 WVNS Development Of A Cement Waste Form For Decontaminated Tank 8D-2 Sludge Wash " document. These responses are contained in attachment "A" of this letter and are based on input from Analytical & Process Chemistry (D. J. Fauth, L. E. Michnik, R. A. Palmer), IRTS Process Control Engineering (J. C. Cwynar) and J. Mahoney Vitrification Process Development.

This letter satisfies commitment DW:4410. If you have any questions, please contact the undersign on extension 4382.

P.S. Klanian, Manager
Vitrification Test Group
West Valley Nuclear Services Co., Inc.

EM:91:0035

PSK:psk
cc:

C. Leek
R. Provencher
J.A. Yeazel

EM:91:0035

0254:91:10

ATTACHMENT A

REVIEW COMMENTS ON JANUARY 1991 WVNS DEVELOPMENT OF A CEMENT WASTE FORM FOR DECONTAMINATED TANK 8D-2 SLUDGE WASH

Part I: Clarification of "Technical Position on Waste Form" (TPWF) Revision 1

NRC QUESTION :

TPWF Section C.2.a - Compressive Strength Requirements.

WVNS has proposed to use two-inch cubes made in accordance with ASTM C109 for the laboratory scale specimens instead of cylinders as directed by ASTM C39. We have determined that this is an acceptable alternative, because the waste forms will not have any particles large enough to dominate the compressive strength test results. The waste is a filtered liquid, concentrated by evaporation to 33 wt% solids, and is unlikely to have any particles at all other than small crystals. If the cement is thoroughly mixed, the uncured waste form is expected to be a smooth paste without particles. Care must be taken not to exceed the load application rates specified in Section 9.6.3 of test method ASTM C109.

WVNS RESPONSE :

Agree (with the following reservation). Because of radiological contamination concerns, we are evaluating alternatives to cube sanding as specified in ASTM C109. NRC concurrence will be requested prior to any such change.

NRC QUESTION :

TPWF Section C.3.e - Leach Testing

The plan and the Topical Report should include the projected concentrations of radionuclide tracers, and the basis for selecting these concentrations.

WVNS RESPONSE :

Agree. This test is currently being designed. The information will be supplied after the analytical chemistry method has been developed and approved. This method will be issued by June 14, 1991.

NRC QUESTION :

TPWF Appendix A - Cement Stabilization

Section II.B - Compression

The plan and the Topical Report should specify the number of two-inch cubes that will be prepared for compressive strength testing. The

TPWF recommends a minimum of 10 samples for statistical meaning.

The Plan and the Topical Report should include developing a correlation between the compressive strength data using ASTM C109 compressive strength tests and the penetrometer or other tests to be used in PCP verification testing.

WVNS RESPONSE :

Agree. The Test Request / Procedures, contained in the notebook (previously sent) specify the number of cubes required for each test. Regarding penetrometer tests, the penetrometer data is being used only as an indicator to determine initial set times. Compressive strength tests are being performed on all specimens.

NRC QUESTION :

TPWF Appendix A - Cement Stabilization

Section II.D. - Irradiation

The TPWF recommends irradiation testing under two conditions:

- the specimens contain ion exchange resins or other organic media, or
- the expected cumulative dose is greater than 10E+9 rads.

The sludge wash waste forms will not contain ion exchange media, and the total organic content is expected to be less than 500 ppm. Previous experience with the supernatant cement solidification indicated that oxalates at concentrations less than 39 ug/g appeared to retard the set (Reference 1). However, there is no indication that this concentration of organic would cause the waste form to be vulnerable to irradiation degradation. The sludge wash simulant will be prepared with almost 1% organic acids. Irradiation testing of the supernatant specimens indicated that they were not degraded (Reference 2). Based on this previous data and the proposed sludge wash simulant formulation, we have determined that irradiation testing of the sludge wash qualification specimens is not required.

WVNS RESPONSE :

Agree.

NRC QUESTION :

TPWF Appendix A - Cement Stabilization

Section II.E. - Biodegradation

Biodegradation testing of qualification specimens is recommended by the TPWF only if the waste forms contain carbonaceous materials. As stated above, the sludge wash qualifications specimens will be prepared with 0.5% organic constituents. Since the supernatant

specimens were spiked with more than this amount, and their biodegradation test results were acceptable, the Division will not require biodegradation testing of the supernatant specimens.

WVNS RESPONSE :

Agree.

Part II: Plans for Evaluating Sludge Wash Cement Recipe

NRC COMMENT :

1. The plan should specify how many simulant cylinders will be prepared for correlation with the laboratory sized specimens.

WVNS RESPONSE :

Agree. Per Test Procedure TP-030, Twenty-two cores will be obtained from five full-size drums. These cores will be crushed and the compressive strength will be compared to laboratory sized specimens.

NRC COMMENT :

2. The plan should specify how may 1" x 3" cylinders will be prepared for the leach test. Although neither the TPWF nor ANS 16.1 specifies the number of specimens to be prepared for the leach test, it is suggested that test be performed on more than one specimen.

WVNS RESPONSE :

Agree. This information is included in Test Procedure TP-026. We plan to use three 1" x 3" cylinders for the leach test.

NRC QUESTION :

3. The plan should specify how the confirmation cube will be prepared and whether it is from the sample collected in 1989 or the new composite sample collected in 1991. The plan should specify how the confirmation cube will be cured. The plan should also provide the criterion used to determine whether the confirmation cube compressive strength is acceptable. We believe that this criterion should be that the confirmation cube strength is within the range of compressive strengths of the simulated specimens.

WVNS RESPONSE :

Agree. Confirmation cubes will be prepared based on a 129-inch heel using wash from pre-1991 sampled sludge and 129-inch heel with wash from 1991 sampled sludge wash. Criterion for acceptable compressive strength will be established using data generated for the nominal waste simulant in the multivariate tests performed to Test Procedure TP-028. Curing profiles for these cubes will be derived from data generated from a thermocouple instrumented drum in our 20-drum

simulant run (TR-026).

NRC COMMENT :

4. The plan should describe clearly how the concentration limits and operating limits testing will be done. A list of the recipe variations that will be tested, and how the limits will be selected, should be provided.

   The curing regime should also be clarified. It is unclear how many cubes will be in a "set", and how long they will be oven- and ambient-cured before compression testing. It appears that the oven curing temperature is 88°C; this should be equal to the centerline temperature for the supernatant full-size waste forms.

WVNS RESPONSE :

Agree. Our initial recipe variations and the number of cubes are included in the Test Request TR-028 and Test Procedure TP-028. A thermocouple instrumented drum has been temperature profiled in accordance with SIP 91-01. The peak centerline temperature for this drum was 79°C. The centerline temperature returned to ambient temperature in a total of 90 hours. Our curing cycle for all specimens will include an initial 90 hour oven cure with the remaining curing taking place at ambient temperature. The Test Requests / Procedures, contained in the notebook (previously sent) specify the number of cubes required for each test.

NRC COMMENT :

Comparison of the Simulant and Cement Recipe

The table shows a comparison of the supernatant and sludge wash simulants. Based on the comments addressing item 4 above, it appears that this formulation could change slightly as a result of the concentration limits and operating limits testing.

The major differences between the supernatant simulant and the sludge wash simulant are mostly due to the reduction in solids content from 39% to 33%. Other differences, e.g., the decrease in nitrate and the increase in sulfate are explained by the sludge wash analytical data. However, it is unclear why the organic acids, citric, oxalic, and tartaric, are reduced by a proportion greater than that of the solids. Since these have been demonstrated to cause problems with the waste form set, their concentrations in the simulant should be selected to be conservative.

WVNS RESPONSE :

Agree. We have looked over the recipe and are making minor adjustments. These will be included in the notebook material when finalized. Variations for organic constituents will be included and are included in the parameters of the multivariate testing (TP-038). Variable ranges for the tests are being selected conservatively.

West Valley Demonstration Project

Doc. Number WVNS-TP-025

Revision Number 0

Revision Date 05/2?/91
Engineering Release #2098

TEST PROCEDURE

PROCEDURE FOR DEVELOPMENT OF THE NOMINAL RECIPE FOR CEMENT SOLIDIFICATION OF SLUDGE WASH LIQUIDS

PREPARED BY _____ M. N. Baker
Cognizant Engineer

APPROVED BY _____ D. C. Meess
Cognizant System Design Manager

APPROVED BY _____ D. L. Shugars
Quality Assurance Manager

APPROVED BY _____ D. J. Harward
Radiation and Safety Manager

APPROVED BY _____ J. C. Cwynar
Process Control Engineering

West Valley Nuclear Services Co., Inc.

P.O. Box 191

West Valley, NY 14171-0191

BEL0049:3RM

RECORD OF REVISION

PROCEDURE

If there are changes to the procedure, the revision number increases by one. These changes are indicated in the left margin of the body by an arrow (>) at the beginning of the paragraph that contains a change.

Example:

> The arrow in the margin indicates a change.

Rev. No.	Description of Changes	Revision On Page(s)	Dated
0	Original Issue	All	05/24/91

RECORD OF REVISION (CONTINUATION SHEET)

Rev. No.	Description of Changes	Revision on Page(s)	Dated

PROCEDURE FOR DEVELOPMENT OF THE NOMINAL RECIPE FOR CEMENT SOLIDIFICATION OF SLUDGE WASH LIQUIDS

Rev. 0

1.0 SCOPE

1.1 This work is required to develop a stable waste form for cement solidification of Sludge Wash liquids which exhibits the characteristics required by 10 CFR 61, Code of Federal Regulations, Title 10, "Licensing Requirements for Land Disposal of Radioactive Waste," and the USNRC Branch Technical Position on Waste Form, revision 1, dated January, 1991.

1.2 The recipe for cement solidification of supernatant (DOE/NE/44139-49) will be used as a starting point for this test procedure.

1.3 Work will be performed with a simulant representing the actual waste liquid to develop the "nominal" recipe for solidification of sludge wash liquids.

1.4 A prerequisite for all work will be the decision by the IRTS Restart Task Force as to the actual level of supernatant liquid in High-Level Waste Tank 8D-2: 129-inches.

1.5 A prerequisite for all work will be the determination by the IRTS Restart Task Force of the expected amount of Sulfate in the Sludge Wash Liquid. The composition of the "nominal" recipe, based on letter No. EK:91:0047, is given in table 1.

1.6 Work will be performed using cubes 2" x 2" x 2" cast from a simulant/cement mixture produced in the Analytical Chemistry Lab.

1.7 The "nominal" percent solids (by weight) in the waste liquid will be determined.

1.8 The "nominal" range of Calcium Nitrate recipe enhancer to be blended with Portland Type I cement will be determined.

1.9 The "nominal" water-to-cement ratio will be calculated as follows:

$$W/C = \frac{\text{(Weight of waste) x (1 - solids fraction)}}{\text{(1 - Calcium Nitrate fraction) x (Weight of cement blend)}}$$

1.10 Determine the "nominal" amount of Antifoam recipe enhancer to be added to the liquid mixture.

1.11 Determine the "nominal" amount of Sodium Silicate recipe enhancer to be added to the waste/cement mixture.

1.12 The maximum practical compressive strength of the waste form will be determined.

1.13 The effects of variable recipe parameters on the "gel time" and free liquid of the waste mixture will be evaluated.

1.14 The "nominal" recipe developed herein will be scaled up and qualified in accordance with WVNS-TRQ-026. It will also serve as the "nominal" recipe for Process Control Plan parametric window tests being performed under WVNS-TRQ-028.

2.0 DEFINITIONS AND ABBREVIATIONS

2.1 Definitions

Cement - Dry Portland Type I cement in accordance with ASTM Standard C-150-85.

Cement Blend - A homogenous mixture of Portland Type I cement with a percentage of technical grade flake or granular form calcium nitrate with NO ammonium nitrate.

Cast - A specimen mixed in a poly bottle and then poured into a mold.

Cube - A 2" x 2" x 2" cast specimen.

2.2 Abbreviations

ACM - Analytical Chemistry Method
A&PC - Analytical & Process Chemistry
ACP - Analytical Chemistry Procedure
CSS - Cement Solidification System
IRTS - Integrated Radwaste Treatment System
IWP - Industrial Work Permit
PCE - Process Control Engineering
QA - Quality Assurance
R/S - Radiation & Safety
TDS - Total Dissolved Solids

3.0 RESPONSIBILITIES

3.1 Analytical & Process Chemistry performs all work in this Test Procedure.

3.2 Process Control Engineering (PCE) provides technical direction, and compares the test data to the Test Request.

3.3 Quality Assurance provides surveillance to ensure that the requirements of this test procedure are satisfied, and verifies that portions of the test (where independent verification is required) were performed.

3.4 Radiation & Safety monitors radiation and contamination levels

4.0 TOOLS, EQUIPMENT, COMPONENTS, AND REFERENCES

4.1 Tools and Equipment

- "LIGHTNIN" Model TS-1515 lab mixer with high shear impeller
- 2" x 2" x 2" plastic cube molds
- 500 mL poly bottles
- 250 mL poly bottles
- 20 mL scintillation vial
- magnetic stir plate and stirring bar
- timer
- top-loading analytical balance
- Forney Model FT-40-DR compressive strength testing machine with hydraulic power unit and capping set

4.2 Reagents

- Portland Type I cement per ASTM C-150-85
- General Electric AF9020 antifoam emulsion
- Sodium Silicate solution: Water-based solution with 28.5 to 29.5 percent SiO_2
- Powdered calcium nitrate

4.3 Components

- Despatch Series 16000 Environmental Chamber fully operational

4.4 References

4.4.1 EP-11-001, Test Control

4.4.2 EP-11-003, Development Test Control

4.4.3 WVNS-TPL-70-011, Test Plan for Waste Form for Cement Solidification of Sludge Wash Liquids

4.4.4 WVNS-TRQ-025, Test Request for Development of the Nominal Recipe for Cement Solidification of Sludge Wash Liquids

4.4.5 ACM-CEMPREP-4801, Preparation of Cement Samples in the Radiochemistry Lab written by C. W. McVay, et. al.

4.4.6 ACP 7.2, Safety Practices for the Analytical & Process Chemistry Department

4.4.7 WVDP-010, WVNS Radiological Controls Manual

4.4.8 WVDP-011, WVNS Industrial Hygiene & Safety Manual

4.4.9 USNRC Branch Technical Position on Waste Form, revision 1, draft dated December 1990

4.4.10 ASTM Standard C-109, Standard Test Method for Compressive Strength of Hydraulic Cement Mortars (Using 2-inch or 50-mm cube specimens)

4.4.11 ASTM Standard C-617, Practice for Capping Cylindrical Concrete Specimens

4.4.12 ASTM Standard C-470, Specification for Molds for Forming Concrete Test Cylinders Vertically

4.4.13 ASTM Standard C-150, Specification for Portland Type I Cement

5.0 GENERAL INFORMATION

5.1 Performance of the "nominal" waste form developed under this procedure will be qualified under Test Request WVNS-TRQ-026, and Test Procedure WVNS-TP-026.

5.2 Quality Assurance should be notified prior to commencement of activities, in order to perform surveillance(s).

5.3 OPERATORS SHOULD PERFORM FREQUENT CHECKS ON SYSTEMS THAT ARE TURNED ON OR SHUT DOWN TO ASSURE THAT THE SYSTEM DOES WHAT IS EXPECTED, I.E., WATER FLOWS, PRESSURE RISES, ETC. IF THE REQUIRED ACTION THAT IS SUPPOSED TO HAPPEN DOES NOT HAPPEN, (1) STOP - DO NOT PERFORM THE NEXT STEP, (2) SECURE THE SYSTEM IN A SAFE MODE, AND (3) NOTIFY THE COGNIZANT A&PC SCIENTIST OR COGNIZANT ENGINEER IMMEDIATELY.

6.0 EMERGENCY RESPONSE

6.1 For emergencies in the A&PC Lab, responses will be as directed by ACP 7.2 and WVDP-011.

6.2 For emergencies elsewhere in the plant, responses will be as directed by WVDP-010 and WVDP-011.

7.0 DETERMINATION OF THE "NOMINAL" PERCENT SOLIDS IN THE WASTE

This determination will be made by the IRTS Startup Task Force prior to beginning this work.

8.0 DETERMINATION OF THE "NOMINAL" RANGE OF CALCIUM NITRATE RECIPE ENHANCER TO BE BLENDED WITH PORTLAND TYPE I CEMENT

8.1 Starting with the original recipe for encapsulation of Decontaminated Supernatant, test the performance of the waste form at varying percentages of Calcium Nitrate in the cement blend.

8.1.1 Prepare a cube using the "nominal" blend ratio of 5.7 percent Calcium Nitrate in accordance with ACM-CEMPREP-4801.

8.1.2 Cure the cube at 88 $\pm$ 5 degrees celsius for 48 hours.

8.1.3 After curing, remove the cube mold.

8.1.4 Sand two (2) opposite cube faces until flat

8.1.5 Place the cube in the hydraulic press, and measure the pressure at the cube yield point. Record the pressure, and perform the compressive strength calculation per ASTM Standard C-109. Record on form WV-2301.

8.2 Increase the Calcium Nitrate percentage to 6 percent, 7 percent, 8 percent, 9 percent, etc., up to 12 percent Calcium Nitrate in the cement blend.

8.2.1 Prepare cubes at each new cement blend ratio in accordance with ACM-CEMPREP-4801.

8.2.2 Cure the cubes, remove the molds, cap and perform compressive strength testing in accordance with section 8.1.2 through 8.1.5 above.

8.2.3 Record the gel time, penetration resistance, and slurry density on form WV-2301.

8.2.4 Record the presence or absence of bleed water on form WV-2301. If present, determine the pH.

8.3 Decrease the percentage of Calcium Nitrate in the cement to 4 percent.

8.3.1 Prepare the cubes at this cement blend ratio in accordance with ACM-CEMPREP-4801.

8.3.2 Cure the cubes, remove the molds, sand two opposite cube faces until flat and perform compressive strength testing in accordance with sections 8.1.2 through 8.1.5 above.

8.3.3 Record the gel time, penetration resistance, and slurry density on form WV-2301.

8.3.4 Record the presence or absence of bleed water on form WV-2301. If present, determine the pH.

9.0 CALCULATION OF THE NOMINAL WATER-TO-CEMENT RATIO

9.1 After the Calcium Nitrate fraction of the cement blend is determined (section 8.0 above), calculate the nominal water-to-cement ratio as follows:

$$W/C = \frac{\text{(weight of waste) x (1 - solids fraction)}}{\text{(1 - Calcium Nitrate fraction) x (weight of cement blend)}}$$

10.0 DETERMINATION OF THE NOMINAL AMOUNT OF ANTIFOAM

10.1 After the nominal Calcium Nitrate fraction in the cement blend and nominal water-to-cement ratio have been determined, the nominal amount of antifoam in the recipe is to be verified.

10.2 With all other recipe parameters remaining the same, or as previously determined in section 8.0 and 9.0 above, prepare a cube in accordance with ACM-CEMPREP-4801.

10.3 Cure the cube at 88 ±5 degrees celsius for 48 hours.

10.4 After curing, remove the cube mold.

10.5 Sand two opposite faces of the cube until flat.

10.6 Place the capped cube in the hydraulic press, and measure the pressure at the cube yield point. Record the pressure, and perform the compressive strength calculation in accordance with ASTM Standard C-109. Record on form WV-2301.

11.0 DETERMINATION OF THE NOMINAL WEIGHT OF SODIUM SILICATE RECIPE ENHANCER TO BE ADDED

11.1 After the Calcium Nitrate fraction in the cement, water-to-cement ratio, and amount of antifoam in the nominal recipe have been determined, the nominal amount of Sodium Silicate additive is to be verified.

11.2 With all other recipe parameters remaining the same, or as previously determined in sections 8.0, 9.0, and 10.0, prepare a cube in accordance with ACM-CEMPREP-4801.

11.3 Cure the cube at 88 ±5 degrees celsius for 48 hours.

11.4 After curing, remove the cube mold.

11.5 Sand two opposite faces of the cube until flat.

11.6 Place the capped cube in the hydraulic press, and measure the pressure at the cube yield point. Record the pressure, and perform the compressive strength calculation per ASTM Standard C-109. Record on form WV-2301.

12.0 DETERMINATION OF THE MAXIMUM COMPRESSIVE STRENGTH

12.1 After the Calcium Nitrate fraction in the cement blend, amount of Antifoam additive, amount of Sodium Silicate, water-to-cement ratio for the nominal recipe have been verified, determine the maximum practical compressive strength of the waste form.

12.2 A mean compressive strength in excess of 500 psi after 28 days curing is desired, as discussed in the Branch Technical Position, appendix A.II.B.

12.3 Sufficient samples shall be prepared to determine the mean compressive strength as well as the standard deviation. A minimum of 10 samples shall be evaluated.

12.4 the compressive strength vs. time will be determined as discussed in WVNS-TP-026, section 7.0.

13.0 CURING

13.1 A curing temperature of 88 $\pm$5 degrees celsius as required for Cement Solidification of Decontaminated Supernatant will be used for initial testing.

13.2 When processing full-scale drums under Work Order 9100084, a drum was equipped with thermocouples and a temperature recorder. The

drum temperature as a function of time was plotted. The effect of curing at this temperature profile will be evaluated as discussed in the Branch Technical Position, appendix A.III.B.

13.3 For this procedure, the samples will be bagged and cured in an oven or temperature-controlled chamber.

13.4 The chamber will be equipped with a calibrated thermometer and temperature readings will be continuously recorded.

13.5 All samples will be kept in sealed containers and/or poly bags during curing and storage, as discussed in the Branch Technical Position, appendix A.III.C. This is intended to simulate the environment in a sealed drum.

14.0 DETERMINATION OF "GEL TIMES"

14.1 For all samples cast in the A&PC Lab, the cube molds will be filled in accordance with ACM-CEMPREP-4801, with a 20 mL scintillation vial filled for each cube.

14.2 Visually check for gelation of the cement/waste product in the scintillation vial.

a. Check for gelation every 5 minutes, and do not disturb the vial between these time intervals.

b. Gelation is a subjective determination; however, gelled product is indicated when the 20 mL scintillation vial can be tipped slowly to a 90 degree orientation, and the cement product will not deform or flow, and will retain a line perpendicular to the horizon.

c. Bleedwater is **NOT** to be interpreted as a sign of incomplete gelation. Estimate the quantity and determine the pH if not reabsorbed after 24 hours.

REVISED SALT CONCENTRATIONS FOR THE "NOMINAL" SIMULANT RECIPE BASED ON 128.5" HEEL

CONSTITUENT	FORMULA	WEIGHT
Sodium Nitrate	$NaNO_3$	286 lbs.
Sodium Nitrite	$NaNO_2$	272 lbs.
Sodium Sulfate	Na_2SO_4	170 lbs.
Sodium Bicarbonate	$NaHCO_3$	*
Potassium Nitrate	KNO_3	17.9 lbs.
Sodium Carbonate	Na_2CO_3	48.4 lbs.
Sodium Hydroxide	NaOH	**10.4 lbs.
Sodium Dichromate, Dihydrate	$Na_2Cr_2O_7.2H_2O$	1590 g
Sodium Chloride	NaCl	1310 g
Sodium Phosphate, Dibasic	Na_2HPO_4	950 g
Sodium Molybdate, Dihydrate	$Na_2MoO_4.2H_2O$	226 g
Sodium Tetraborate, Decahydrate	$Na_2B_4O_7.10H_2O$	122 g
Citric Acid, Anhydrous	$C_6H_8O_7$	165 g
Oxalic Acid, Anhydrous	$C_2H_2O_4$	129 g
Tartaric Acid, Anhydrous	$C_4H_6O_6$	180 g
Water	H_2O	1668.0 lbs.
	TOTAL WEIGHT	2483.1 lbs.
Weight of Solids		815.1 lbs.
Weight Percent Solids		32.83 percent

* Note that Sodium Bicarbonate does NOT appear as $NaHCO_3$ at elevated pH's.

** The Sodium Hydroxide (NaOH) value is an approximation to arrive at a pH of 12.1. This value may vary.

SRC4094/1

West Valley
Nuclear Services Company
Incorporated

WD:91:0725 Steve K.

P.O. Box 191
West Valley, New York 14171-0191

MS-M

July 5, 1991

Mr. T. J. Rowland, Director
West Valley Project Office
U. S. Department of Energy
MS-DOE
P. O. Box 191
West Valley, New York 14171-0191

Dear Mr. Rowland:

Attention: W. S. Kotola

SUBJECT: Meeting Minutes for Cement Waste Form Discussion, NRC Site Visit of June 6, 1991.

During the subject meeting, an agreement was reached for WVNS to record meeting minutes and forward a copy to the NRC through the WVPO. Attached are the preliminary meeting minutes. The minutes are termed preliminary until concurrence is achieved with the NRC. Please forward a copy of the minutes to the NRC for review and concurrence.

Very truly yours,

J. L. Mahoney
Acting IRTS Engineering Process Engineer
West Valley Nuclear Services Co., Inc.

D. C. Meess, Manager
IRTS Engineering
West Valley Nuclear Services Co., Inc.

CJ:91:0067

JLM:src

Attachment: A) Preliminary Meeting Minutes for Cement Waste Form Discussion NRC/WVNS Meeting June 6, 1991

SRC4168

0894:91:08

ATTACHMENT A

PRELIMINARY MEETING MINUTES CEMENT WASTE FORM
DISCUSSION NRC/WVNS MEETING JUNE 6, 1991

- Discussed the WVNS thought to have the ASTM committee address the use of templates when performing cube compression testing. The current method only addresses sanding the surfaces as an accepted handling option.

- Acknowledged the update on curing conditions from the instrumented drum produced in the 20-drum run: 79 °C for a total of 90 hours.

- Indicated the radionuclides of interest that will be specifically tested during the leachant-selection test will be Cs-137, Sr-90, Tc-99, and Pu-238/239/240/241. The WVNS plan is to use actual decontaminated supernate left from the operation of STS and to add sufficient quantities of sodium sulfate and sodium hydroxide to simulate sludge wash solution.

- Confirmed the minimum number of samples, per 5.10 of TPL-70-11 was 3 with the exception of 10 for compressive strength.

- Addressed the use of a penetrometer in qualification testing and the development of a correlation to key cement properties. WVNS emphasized that all qualification work was based on compressive strength testing of cured product so no correlation to penetrometer readings is planned. The IRTS Operations staff will probably use a penetrometer reading as an internal guide to the processability and acceptance of individual batches.

- WVNS affirmed the word "verify" in section 1.4 of SIP 91-01 should actually be "develop" as it pertains to the relationship between cube and core compressive strengths.

- Table 6.2.1 of SIP 91-01 was indicated as being out of date. Test Exception (TE) TE-SIP-91-01-06 corrected the names, formulas, and level of minor species (including organics) in the simulant. This led to a general discussion of the issue of forwarding TE's to the NRC. It was agreed that the revised base document, which will fold all outstanding TE's into the updated document, will be forwarded to Steve Keto[illegible], WVPC. This will be accompanied by a one-page write-up that details the underlying causes that spawned the TE's.

- Selection of the parameters for analysis, as noted in 6.2.3 of SIP 91-01 is not specific. Mary Adams felt that some input from the NRC could be forwarded in this area. WVNS indicated that this work is essentially complete and that any alterations or additions cannot be completed.

SRC4168

- Although table 1 of TRQ-025 does not reflect the updated simulant composition (to agree with the update to SIP 91-01), it is not planned to update the table. This reflects the fact that this preliminary recipe development work is complete, also that the test conclusion that the nominal recipe (from supernatant processing) is acceptable for continued evaluation would not change if the experiments were repeated at a slightly different simulant recipe.

- Mary Adams, NRC, wondered about the relationship between the WVNS Analytical Chemistry Methods (ACM's) and the ASTM and ANSI test methods. WVNS explained that the ACM's are explicit instructions to the laboratory technicians on the steps to execute the reference tests (ASTM or ANSI) in the WVNS lab.

- It was agreed that the 4.3.15 of TP-026 would be updated to reflect the correct test request.

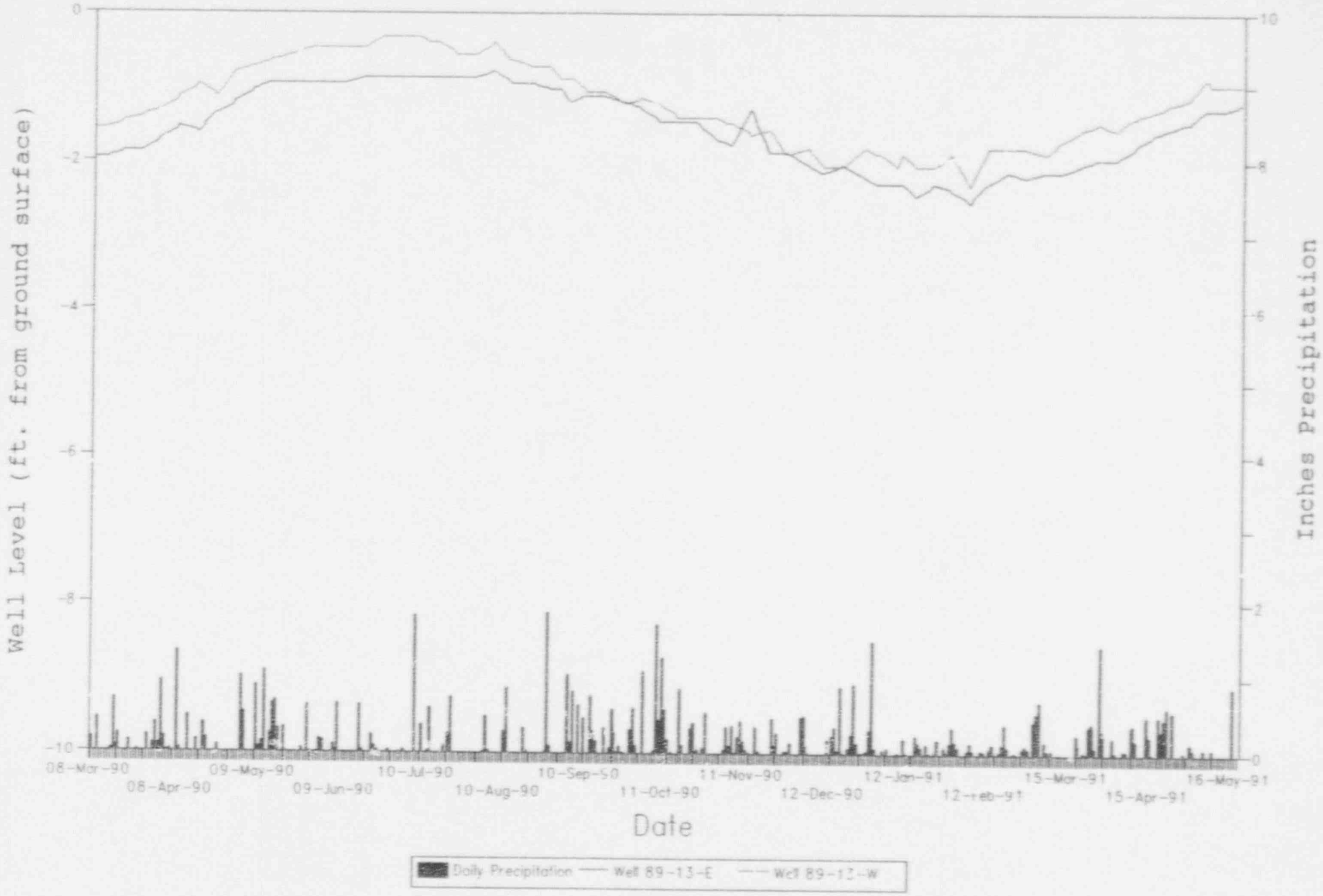
- Sections 7.0 through 7.3, and section 14.0 of TP-026 will be reviewed by WVNS to verify the cylinder and cube totals specified match the values presented in table 2 of that same procedure.

- A pre-release copy of Test Procedure WVNS-TP-025 was forwarded to Mary Adams at the meeting, along with a pre-release copy of the WVNS responses to the questions raised in the 4/22/91 letter from the NRC to WVPO. Each individual response was reviewed. The use of a penetrometer was raised again. WVNS will clarify its use in the overall cement qualification and production process. The test request identified as -038 in the response letter will be updated to -028.

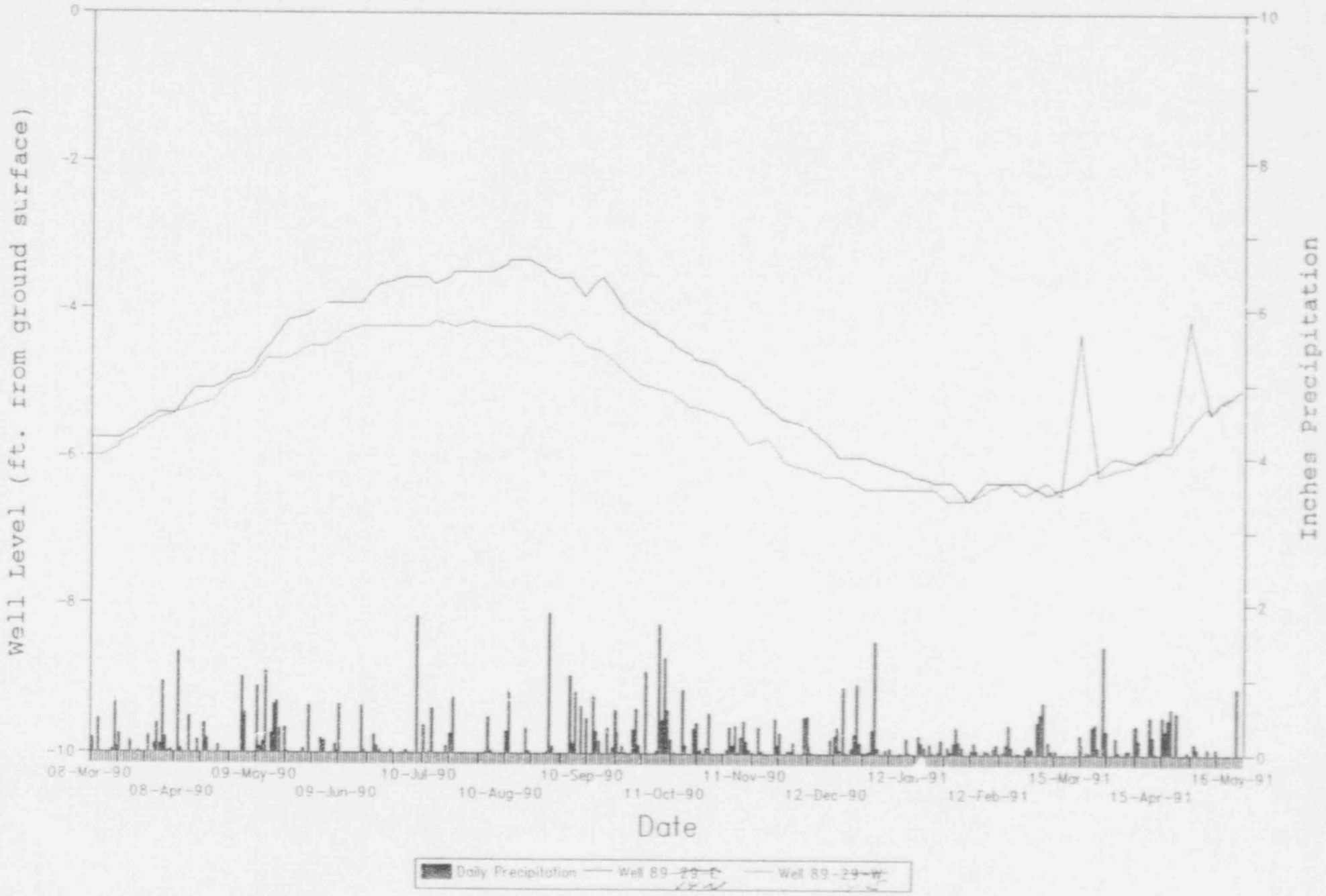
- Discussed some of the long-term testing of supernatant cement that WVNS has begun. Mary Adams clarified the definition of "certain" waste forms (high organics, and/or ion-exchange beads) that require long-term surveillance.

- Tentatively agreed that the NRC TER can be performed from the Test Summary Reports (TSR's) that WVNS is scheduled to generate from the numerous tests. Also felt that a draft copy of the structure of the updated CSS PCP would be helpful to the NRC.

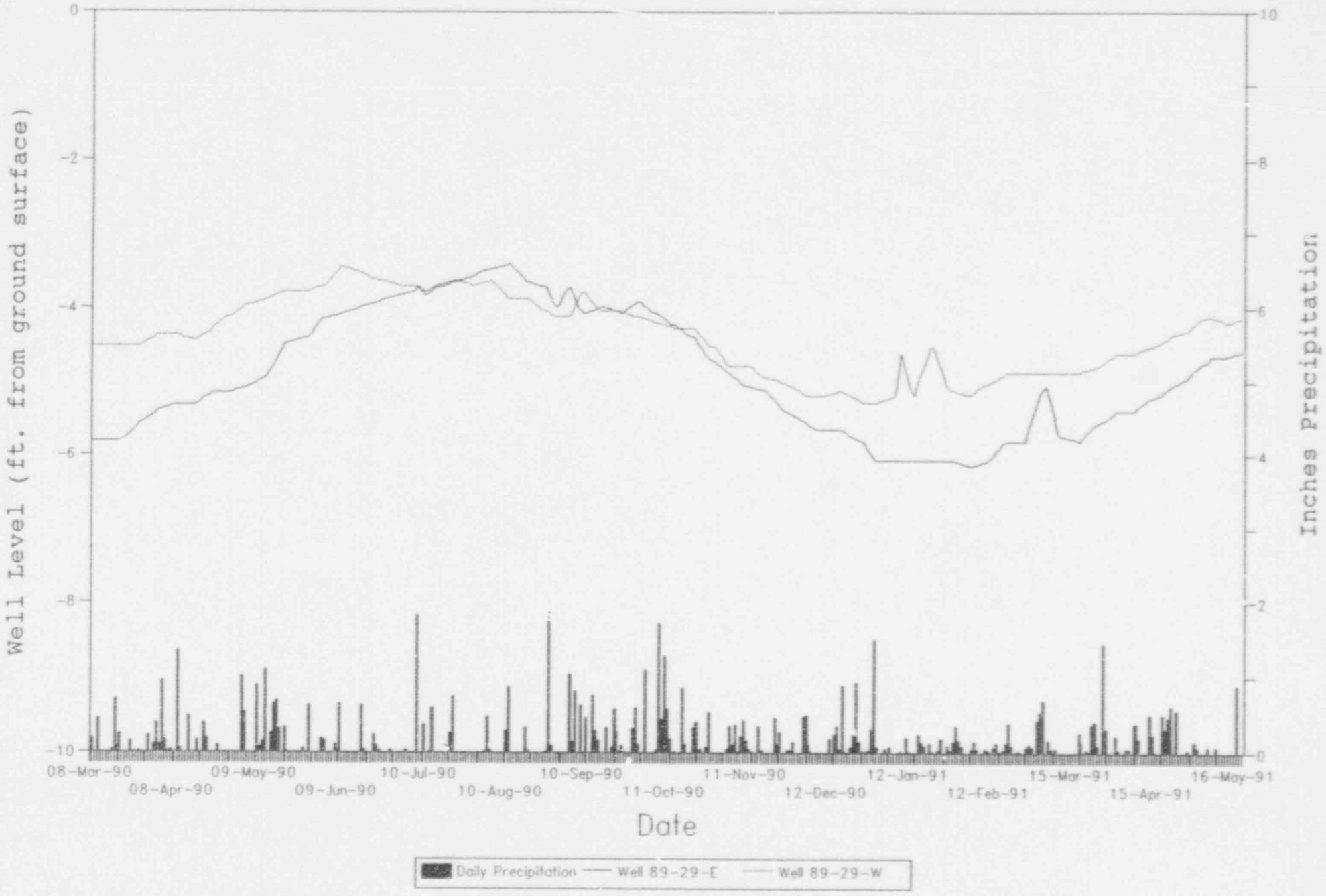
Wells 89-13-E & 13-W Water Level & Daily Precipitation vs. Time



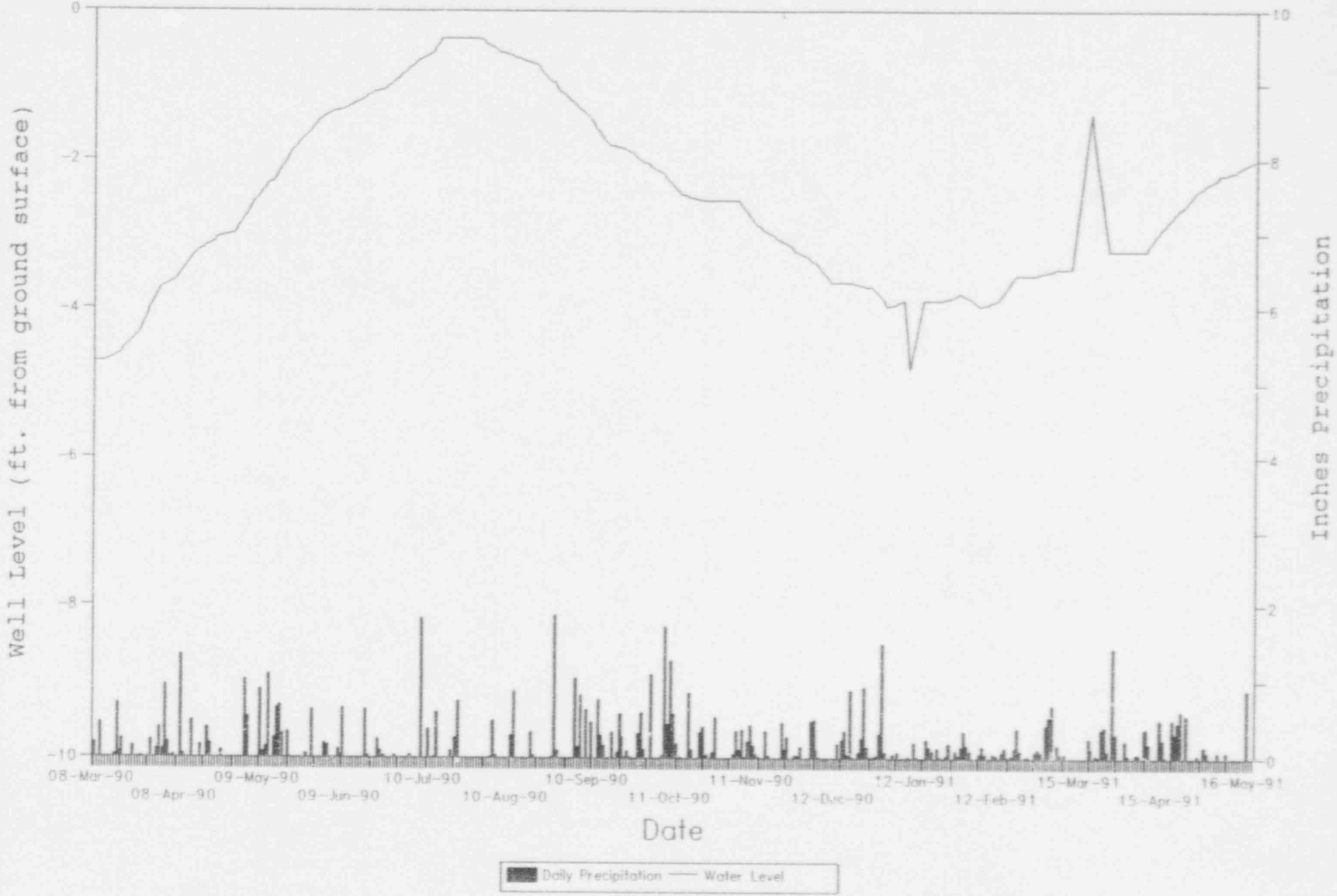
Wells 89-14-E & 14-N Water Level & Daily Precipitation vs. Time



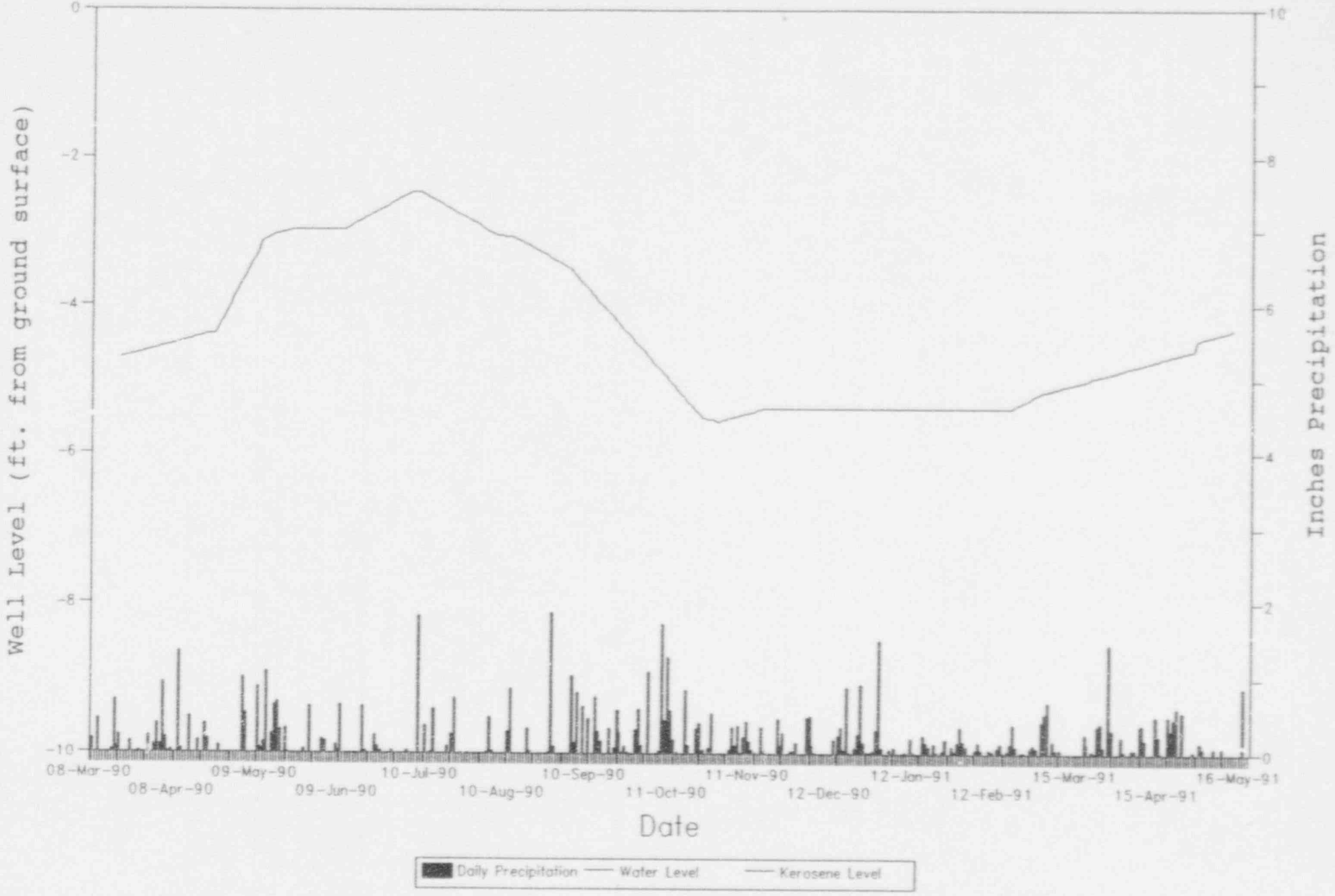
Wells 89-29-E & 29-W Water Level & Daily Precipitation vs. Time



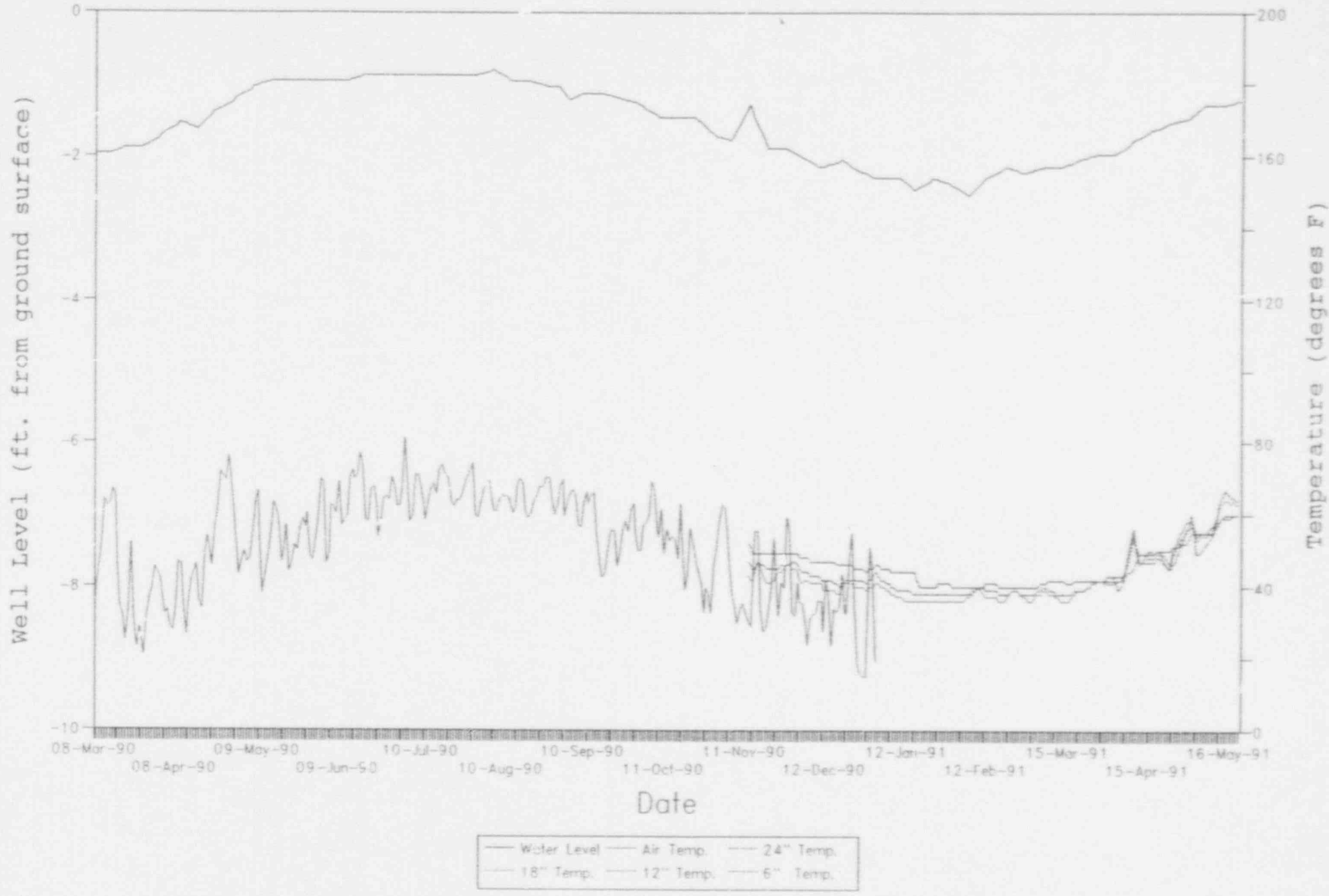
Well 89-05-N Water & Daily Precipitation vs. Time



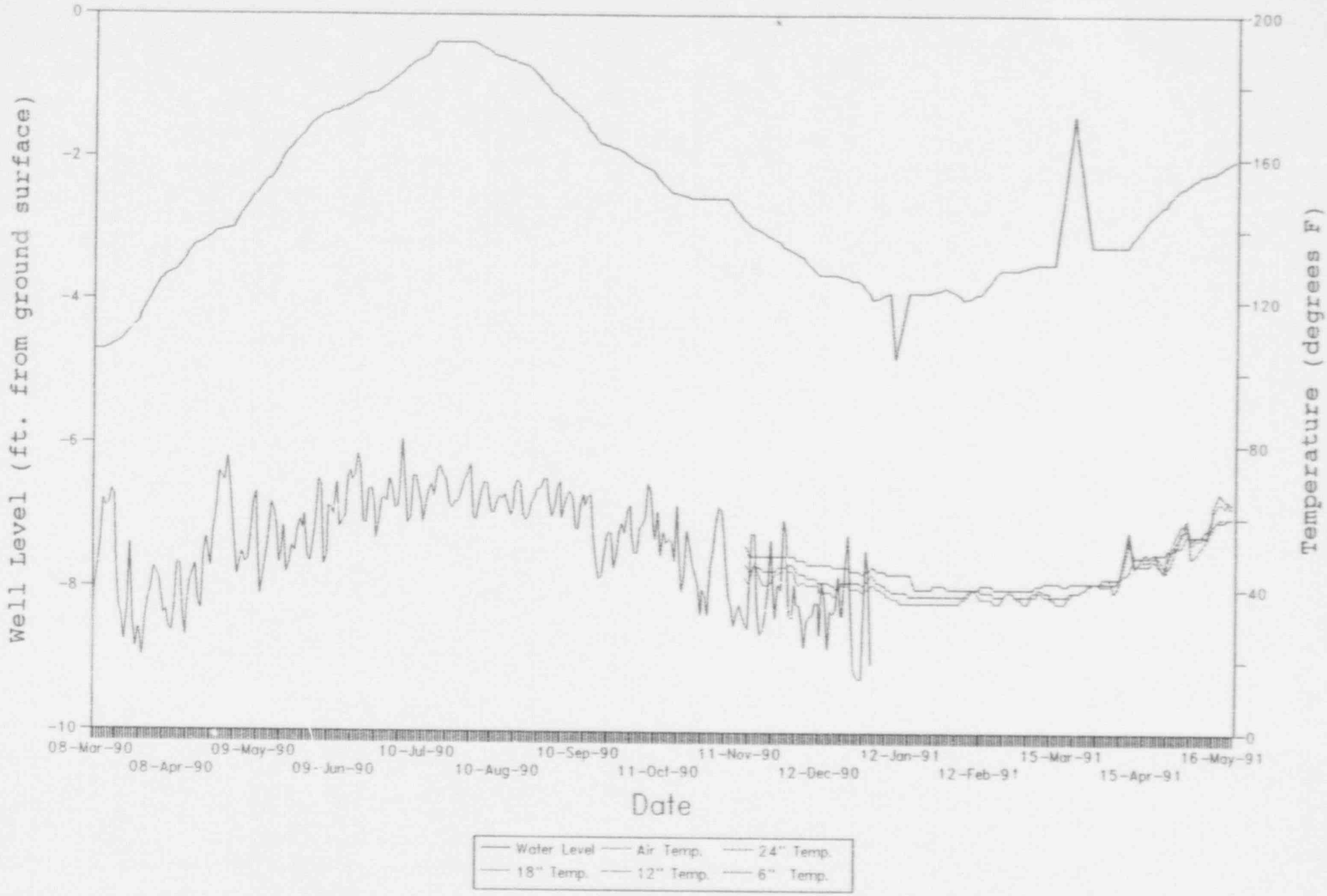
Well 89-28-W Water & Kerosene and Daily Precipitation vs. Time



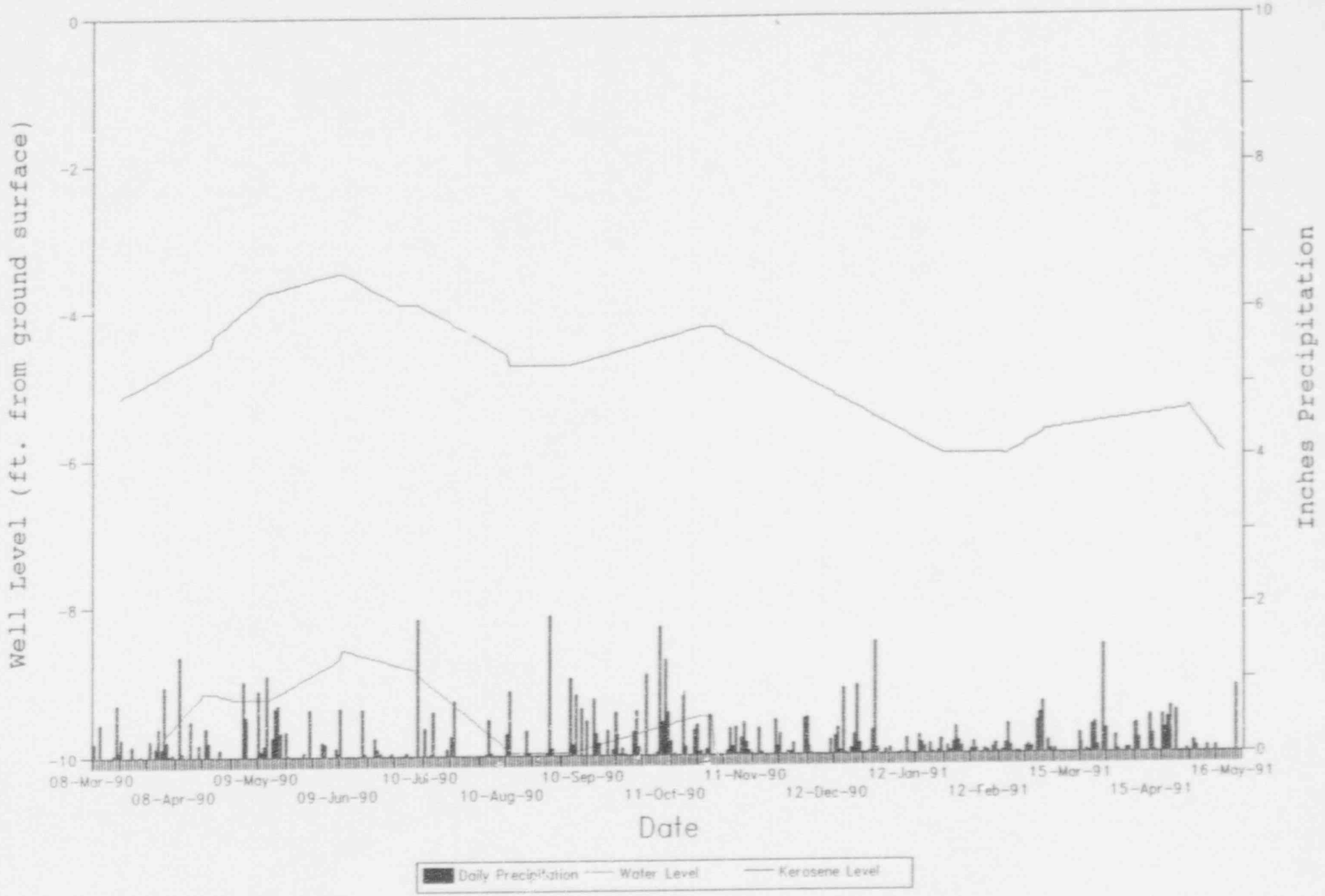
Well 89-13-E Water Level & Air and Ground Temps. vs. Time



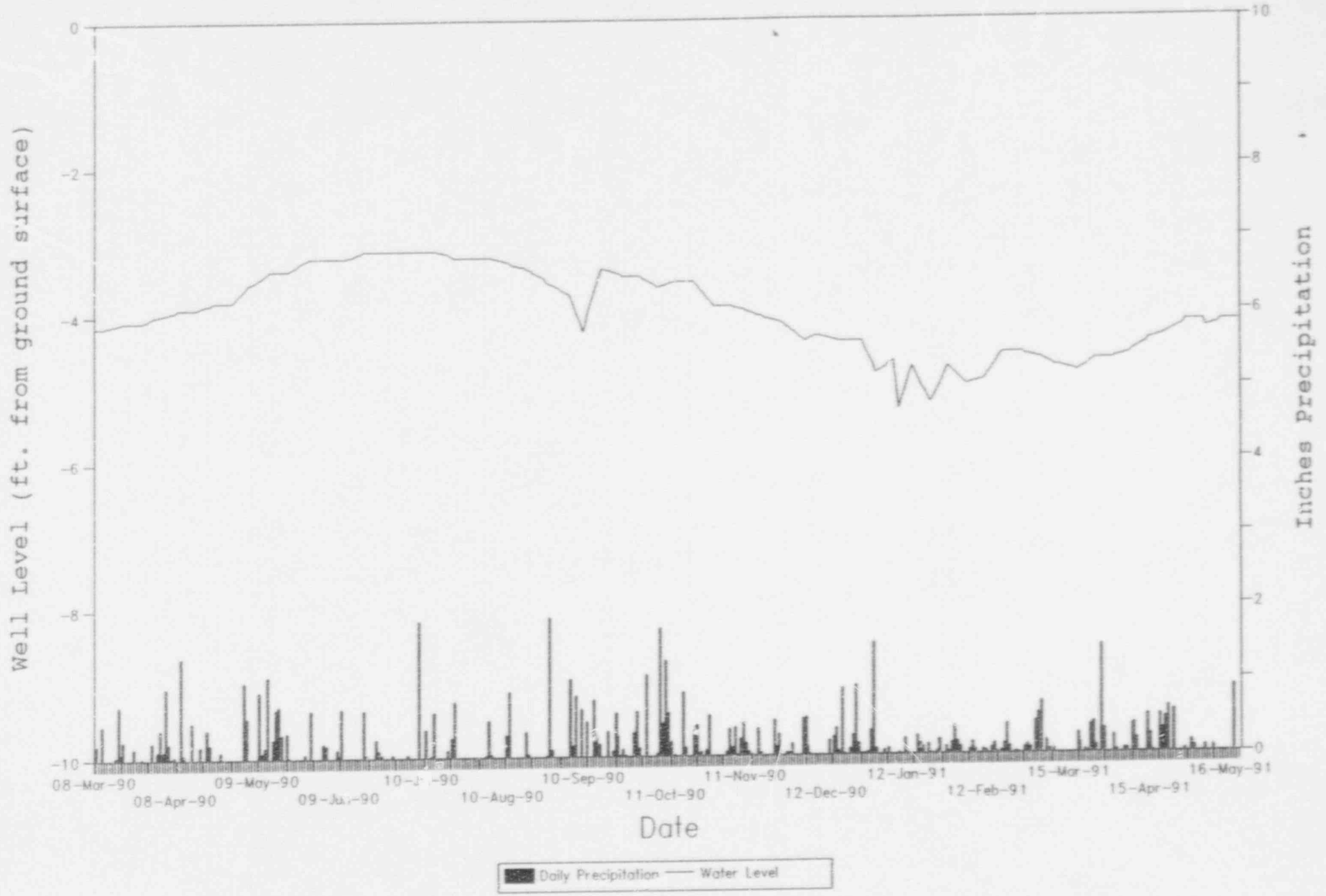
Well 89-05-N Water Level & Air and Ground Temps. vs. Time

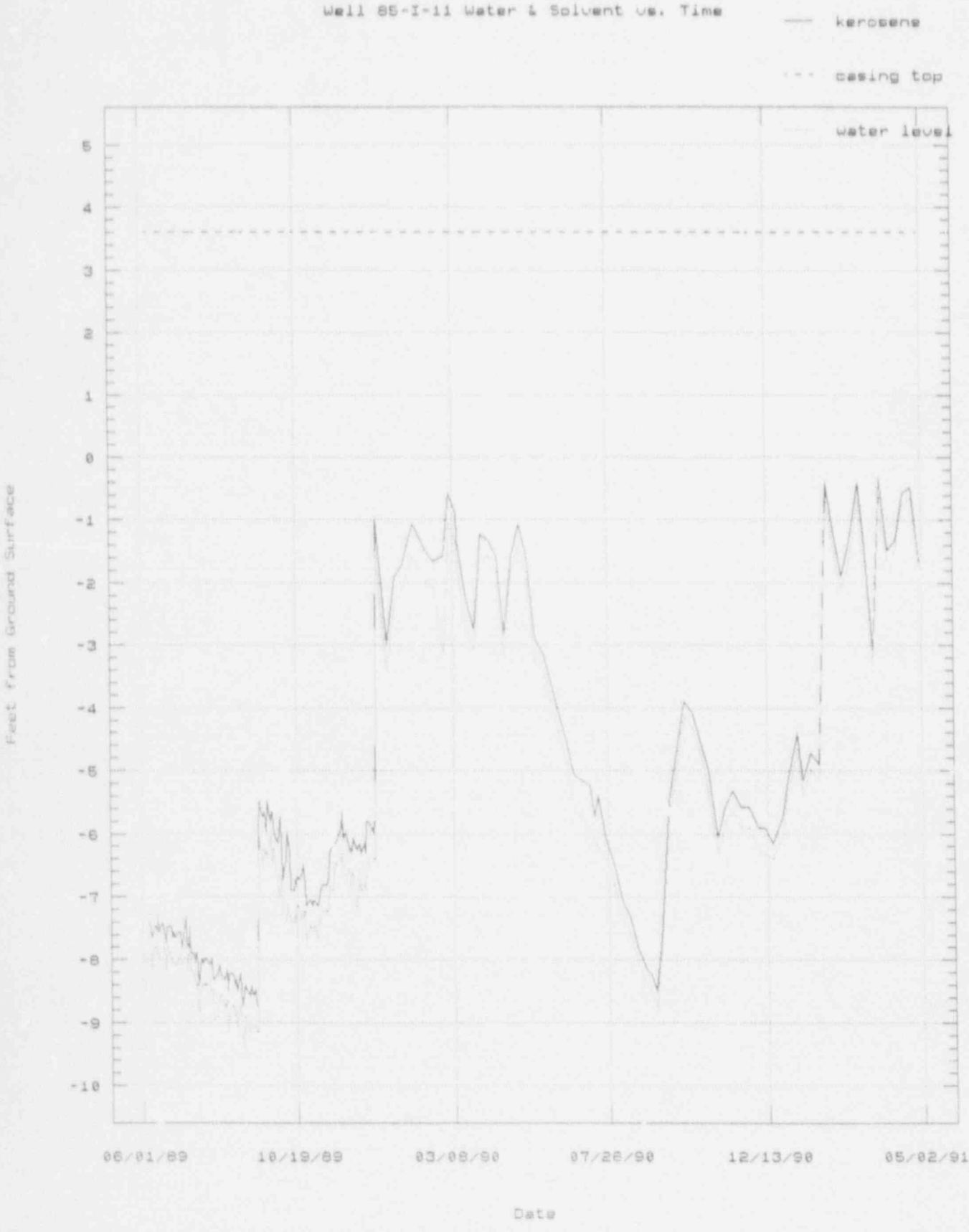


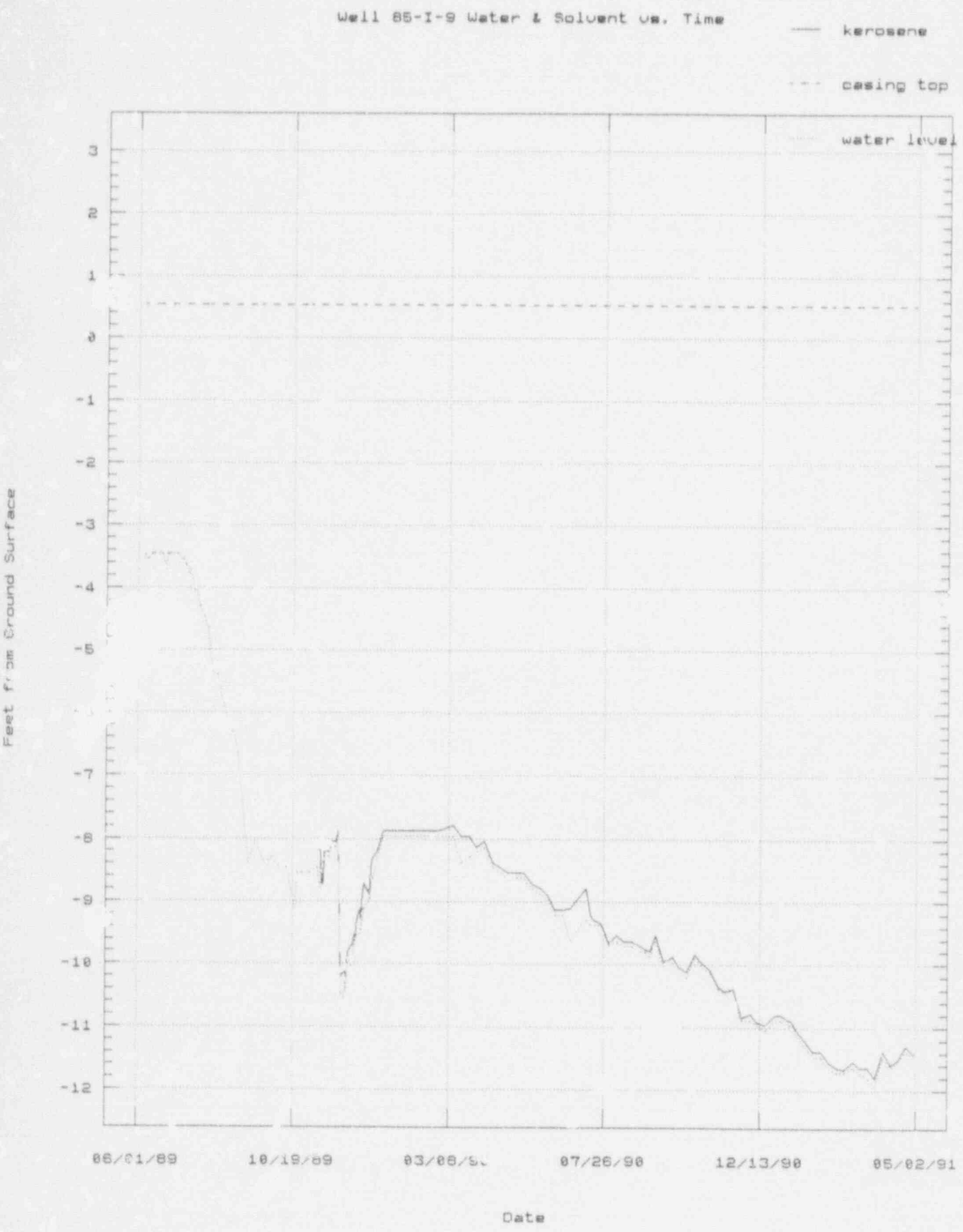
Well 89-27-W Water & Kerosene and Daily Precipitation vs. Time

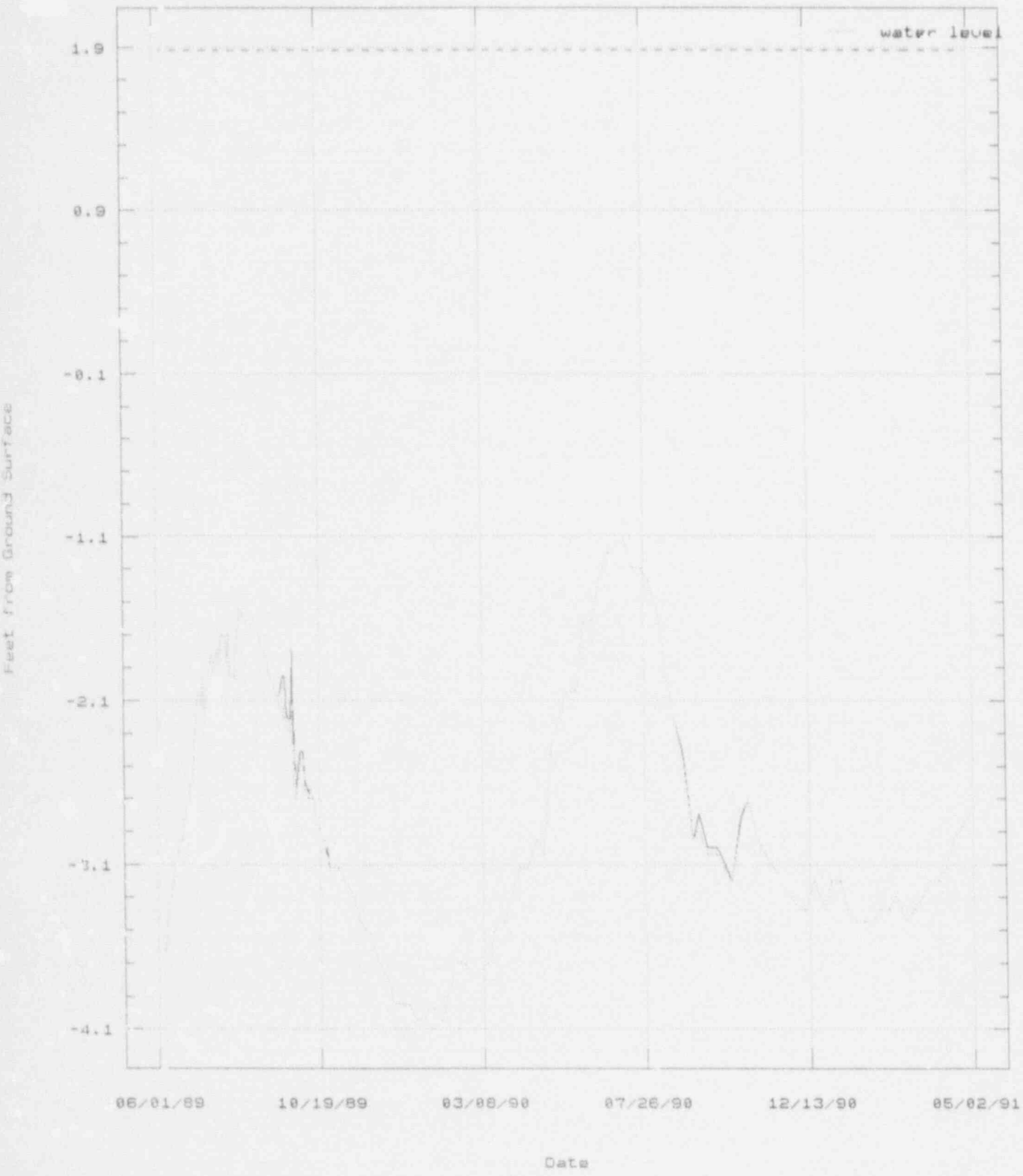


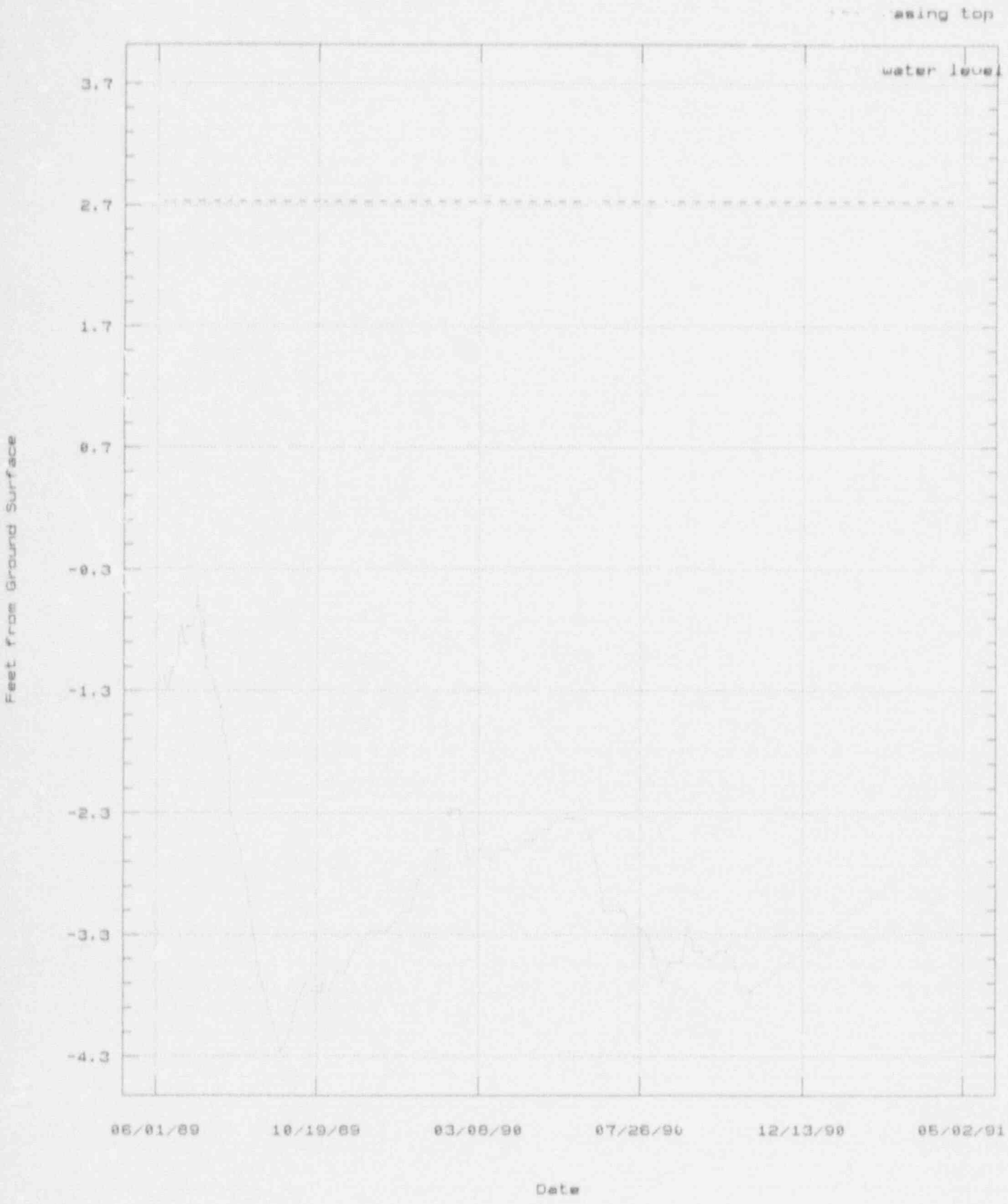
Well 89-27-N Water & Daily Precipitation vs. Time




Well 85-I-11 Water & Solvent vs. Time
kerosene
casing top
water level
5
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 85-I-9 Water & Solvent vs. Time
kerosene
casing top
water level
3
2
1
0
-1
-2
-3
-4
-5
-7
-8
-9
-10
-11
-12
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 86-I-i Water & Solvent vs. Time
kerosene
casing top
water level
1.9
0.9
-0.1
-1.1
-2.1
-3.1
-4.1
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 86-I-3 Water Level vs. Time
asing top
water level
3.7
2.7
1.7
0.7
-0.3
-1.3
-2.3
-3.3
-4.3
Feet from Ground Surface
06/01/89
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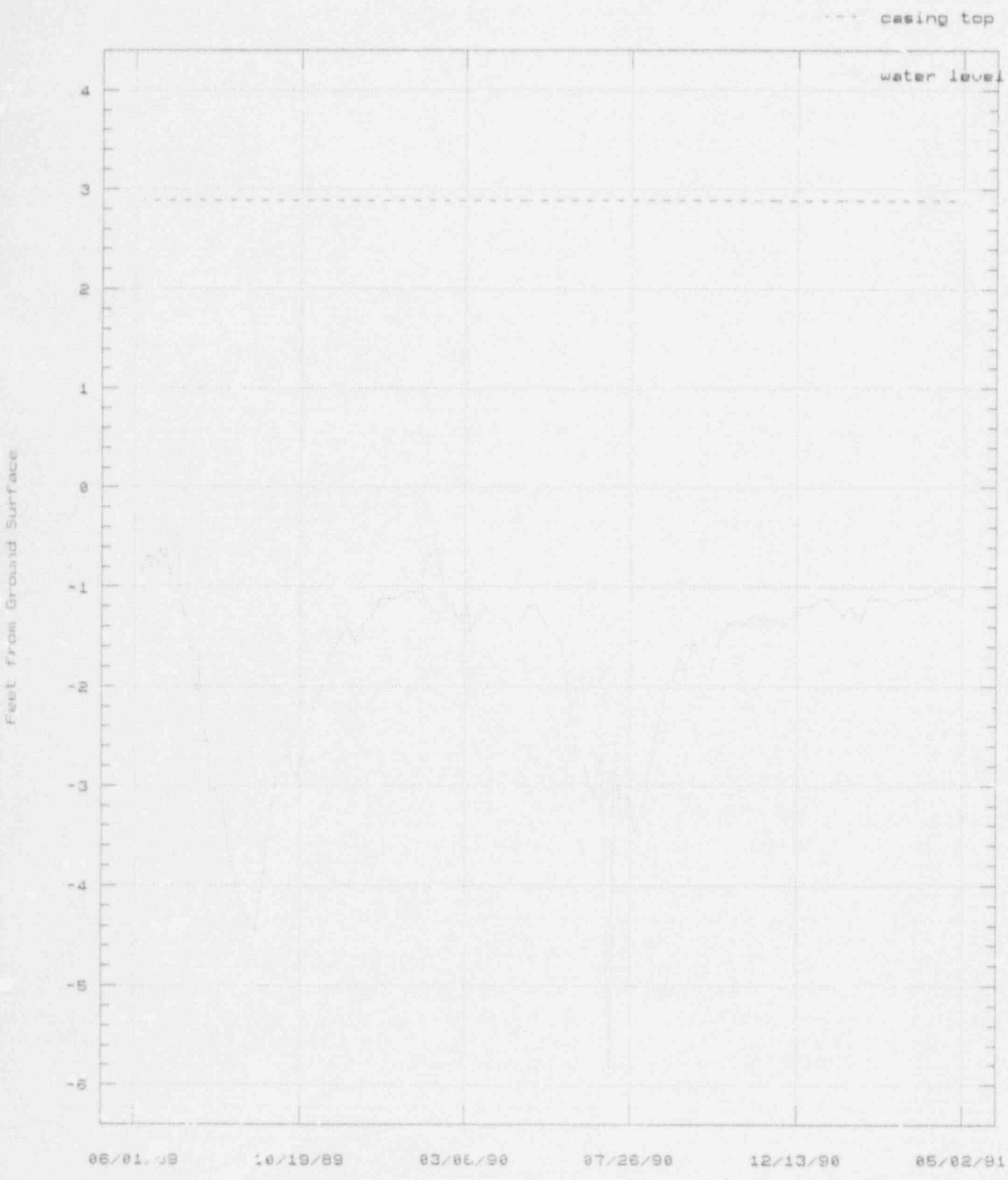
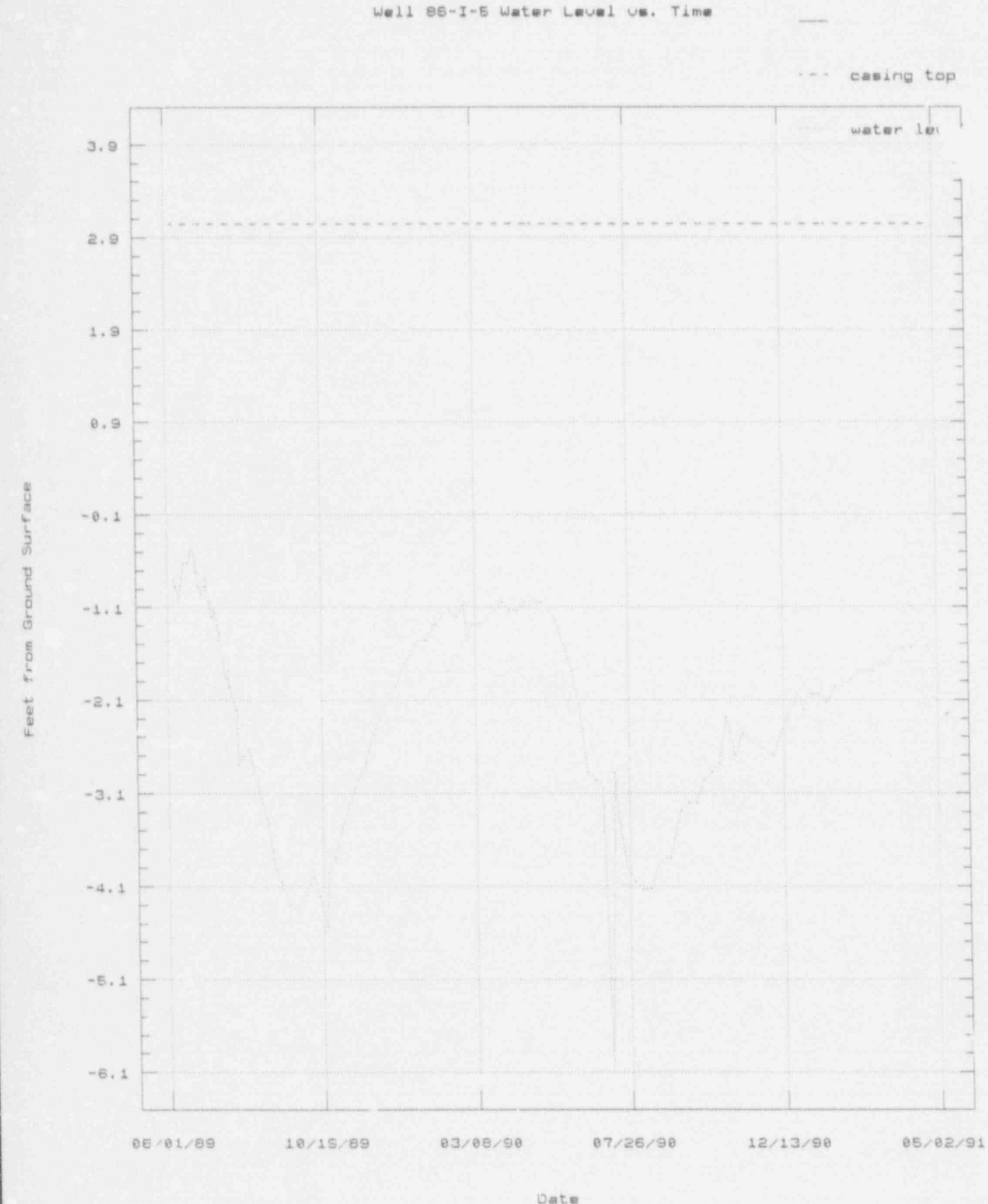
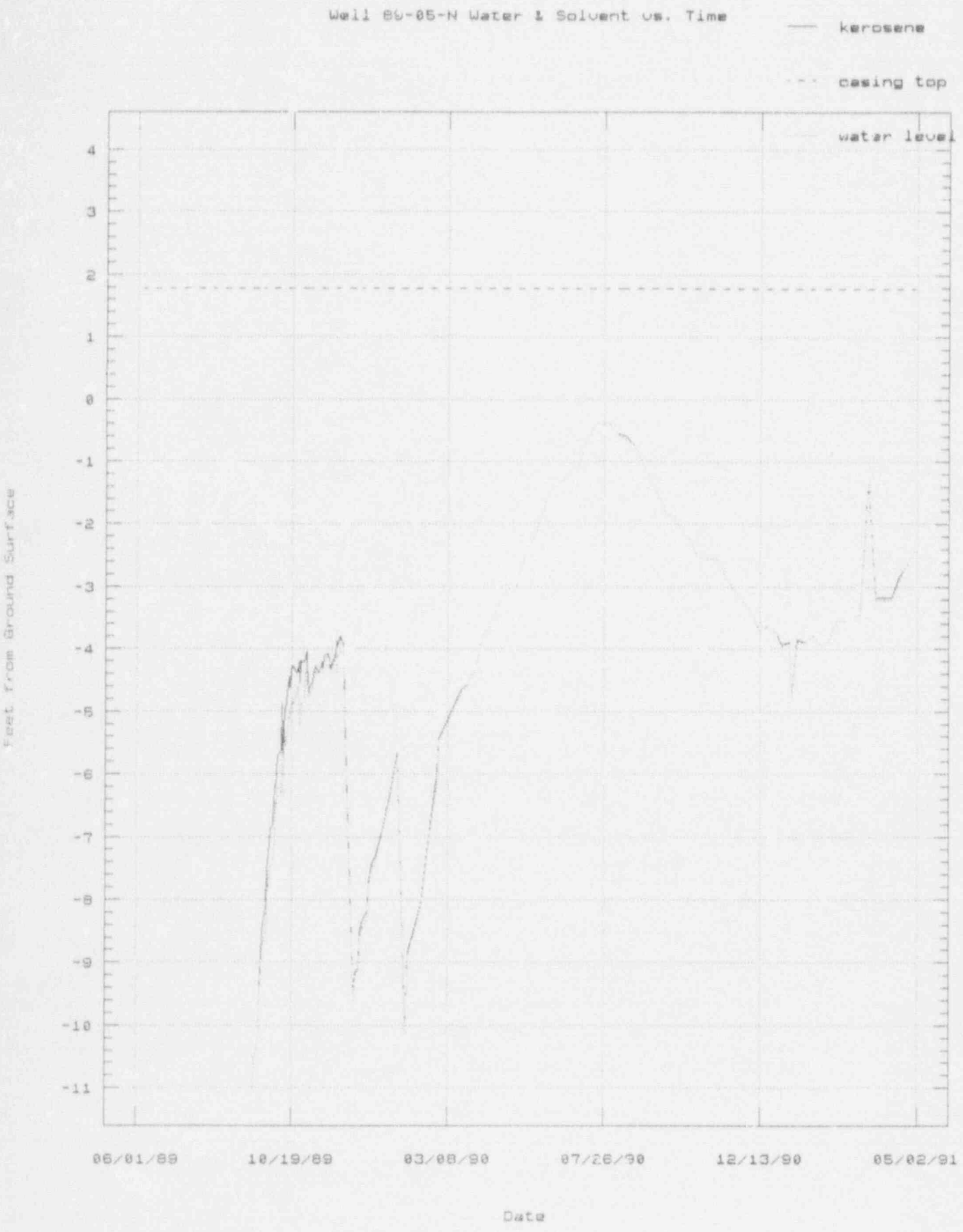
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 86-I-4 Water Level vs. Time
casing top
water level
4
3
2
1
0
-1
-2
-3
-4
-5
-6
Feet from Ground Surface
06/01/89
10/19/89
03/06/90
07/26/90
12/13/90
05/02/91
Date


Well 86-I-5 Water Level vs. Time
casing top
water le
3.9
2.9
1.9
0.9
-0.1
-1.1
-2.1
-3.1
-4.1
-5.1
-6.1
Feet from Ground Surface
06/01/89
10/15/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 86-05-N Water & Solvent vs. Time
kerosene
casing top
water level
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
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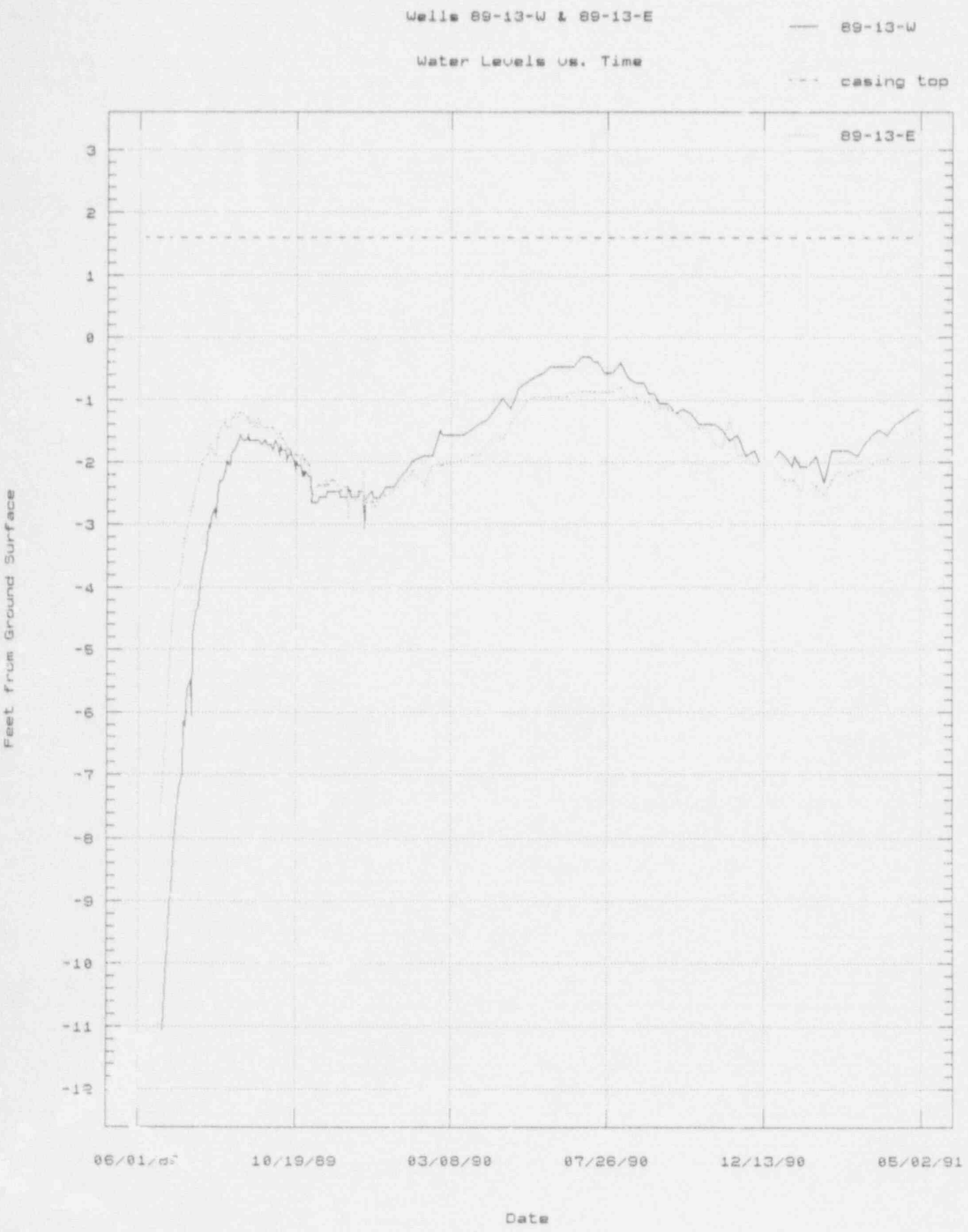
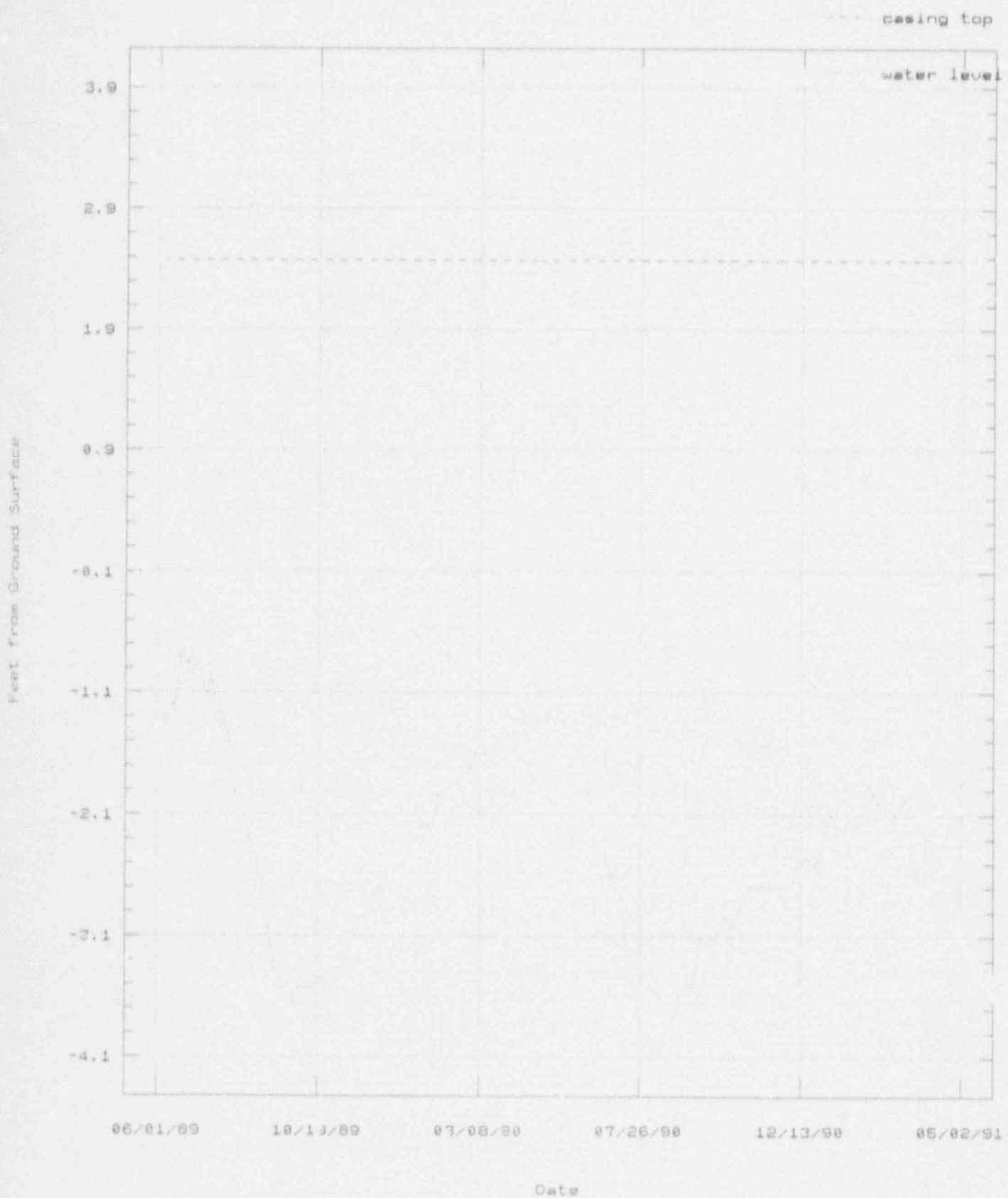
07/26/90
12/13/90
05/02/91
Date


Wells 89-13-W & 89-13-E
Water Levels vs. Time
89-13-W
casing top
89-13-E
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 86-I-6 Water Level vs. Time
casing top
water level
3.9
2.9
1.9
0.9
-0.1
-1.1
-2.1
-3.1
-4.1
Feet from Ground Surface
06/01/89
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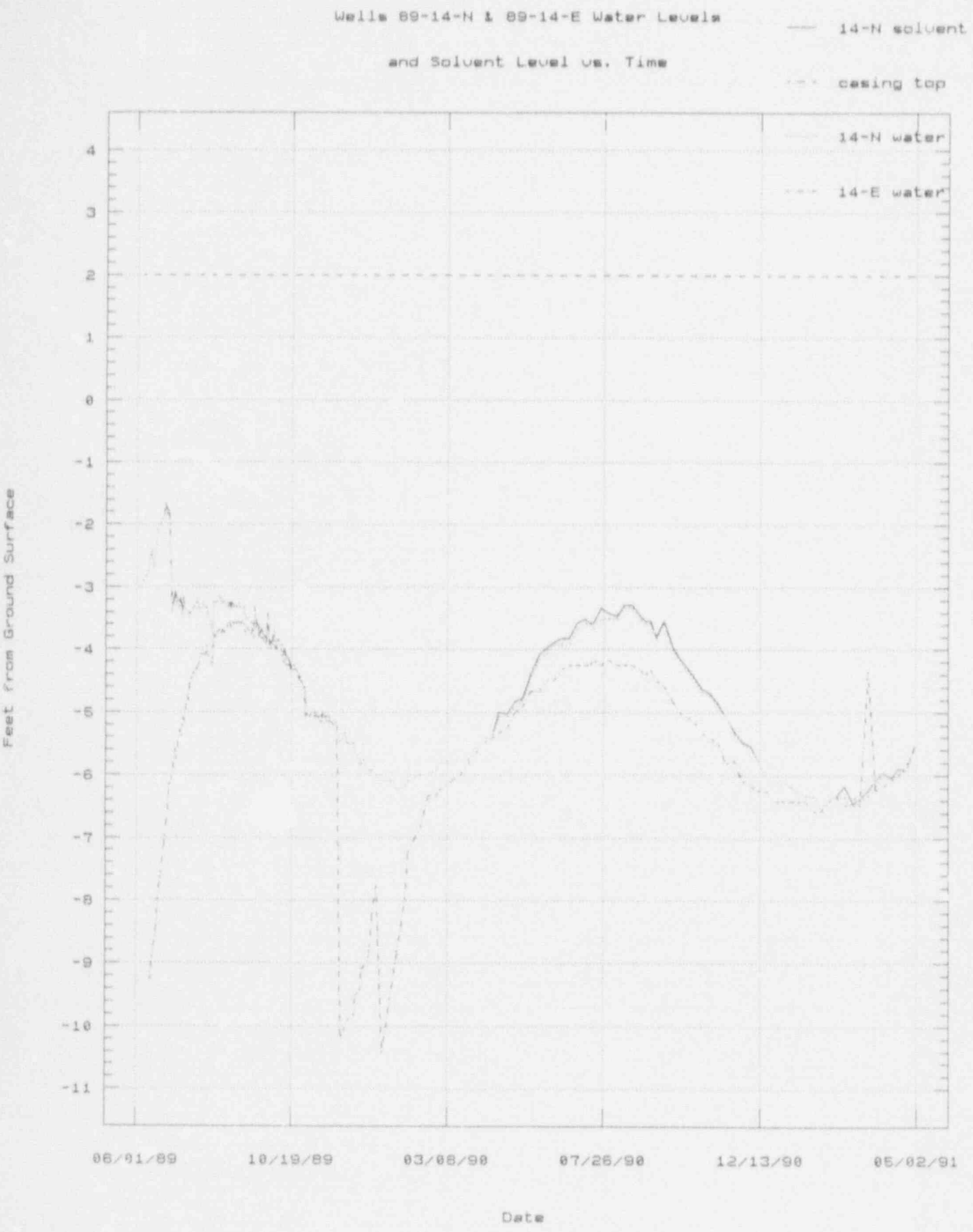
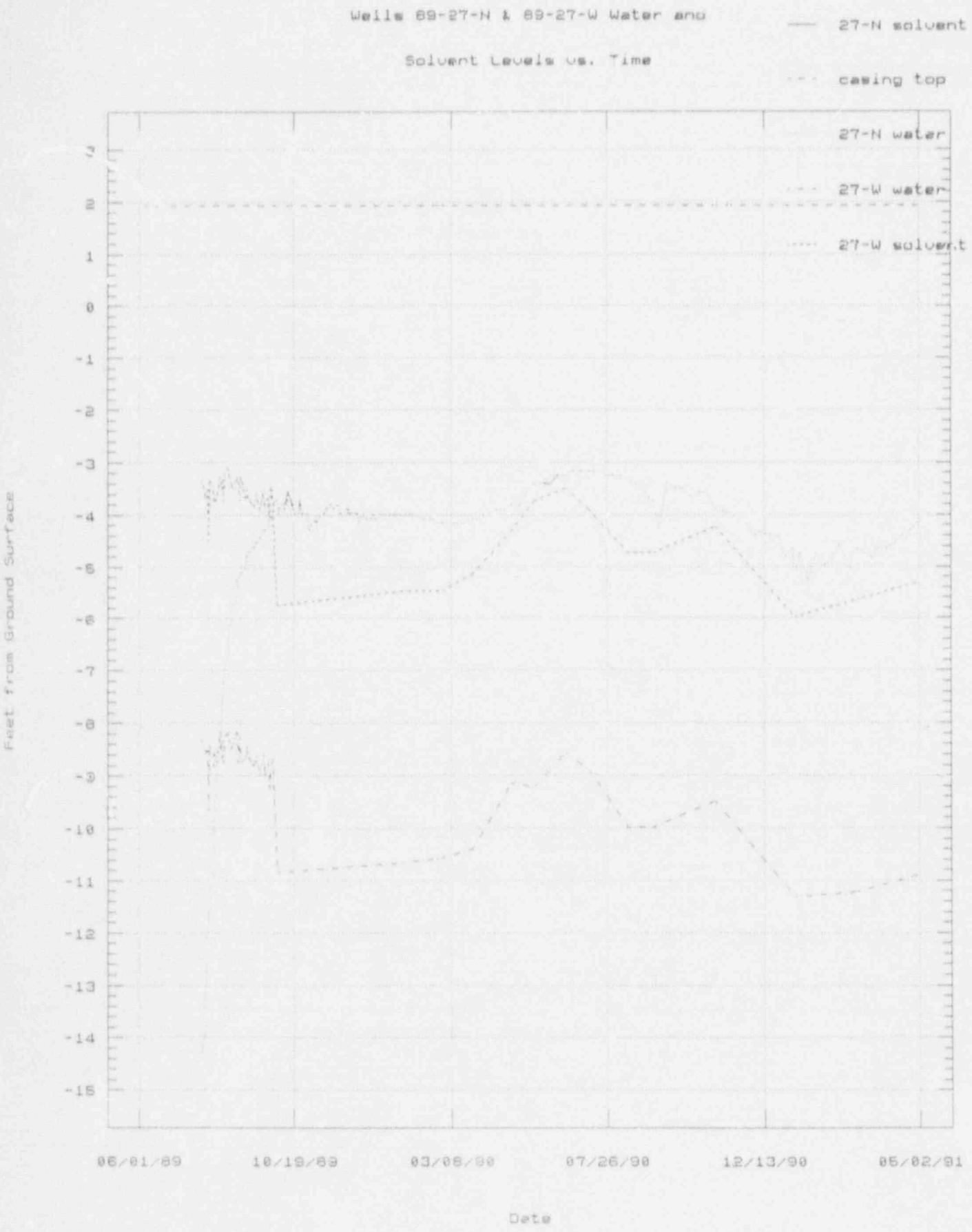
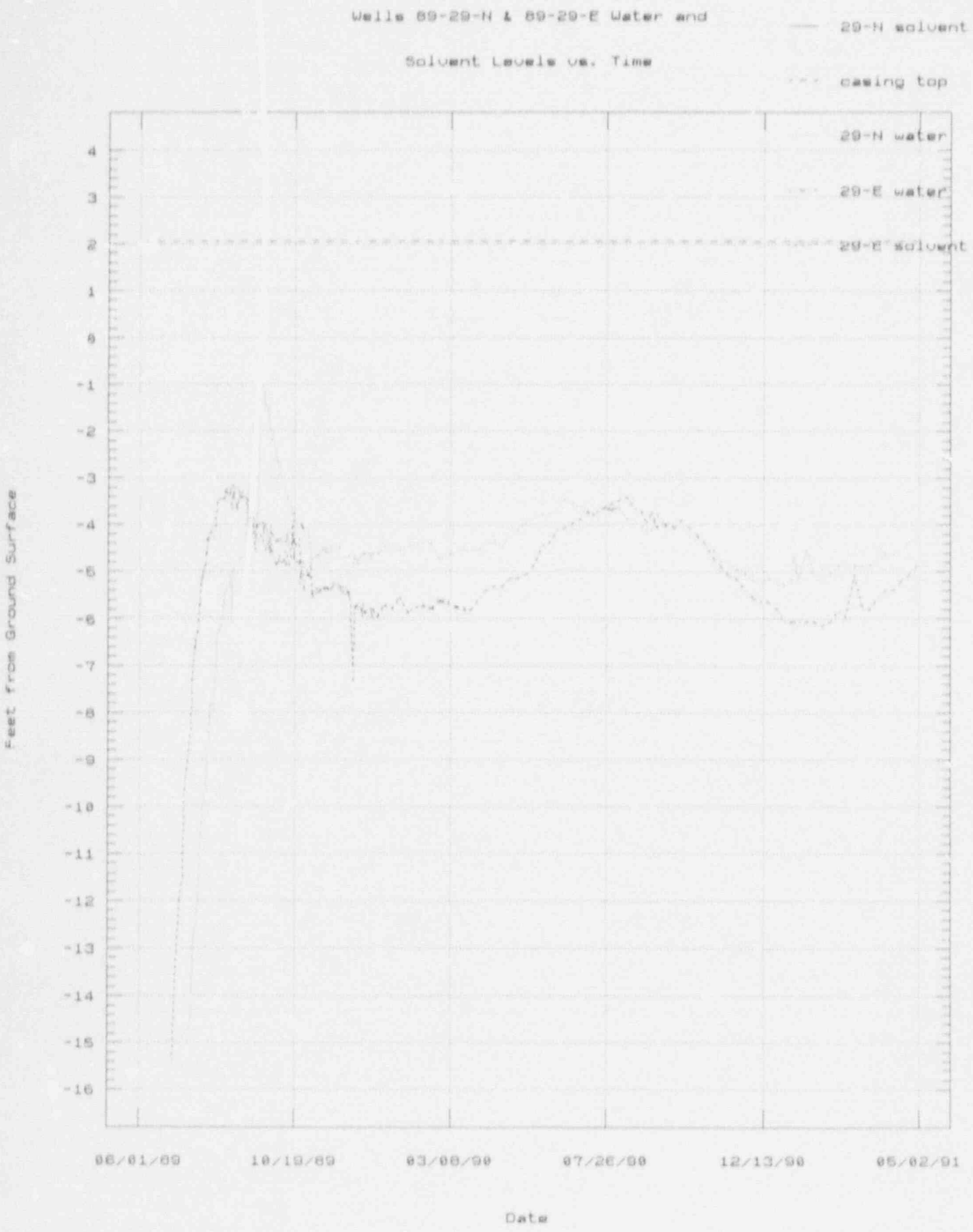
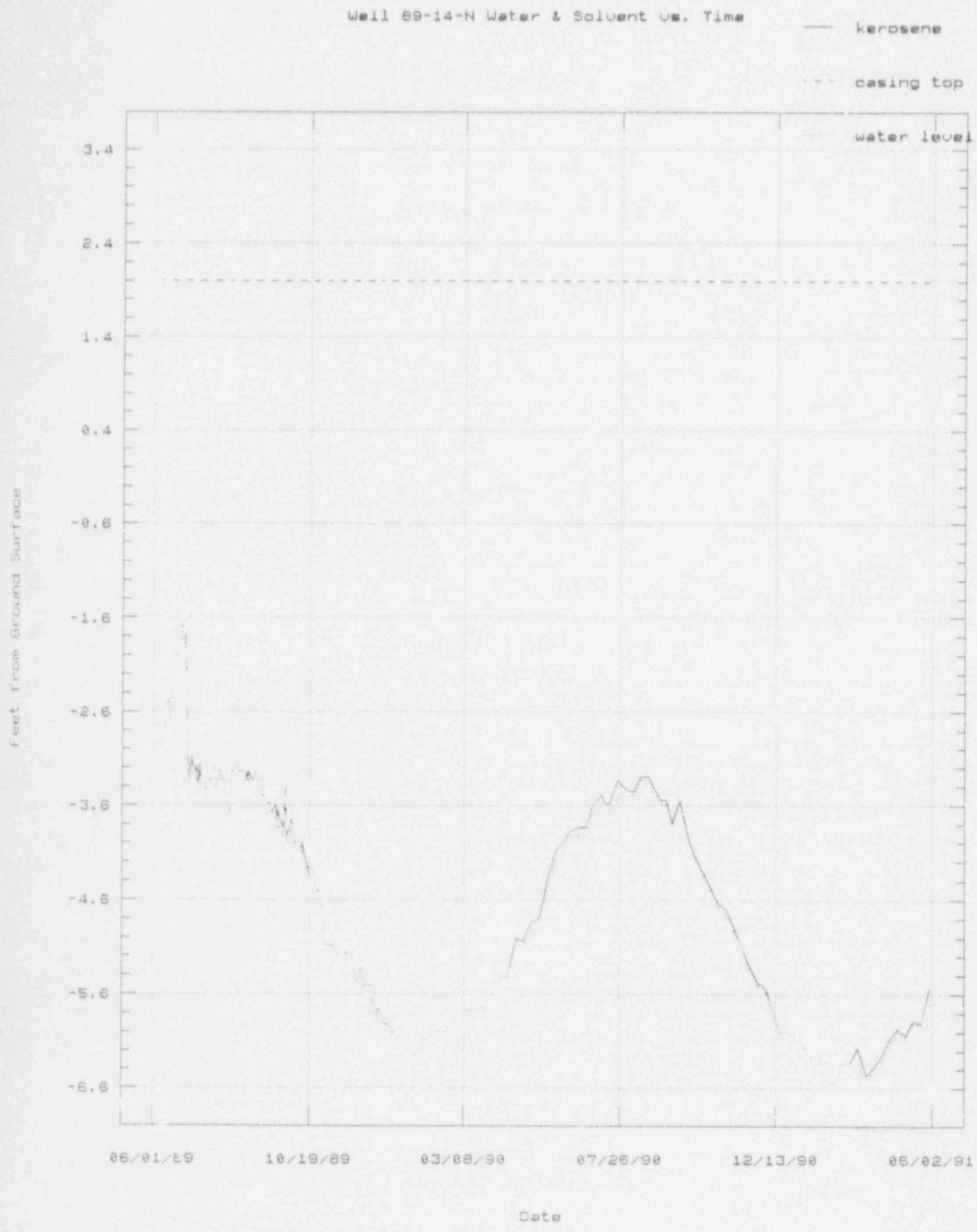
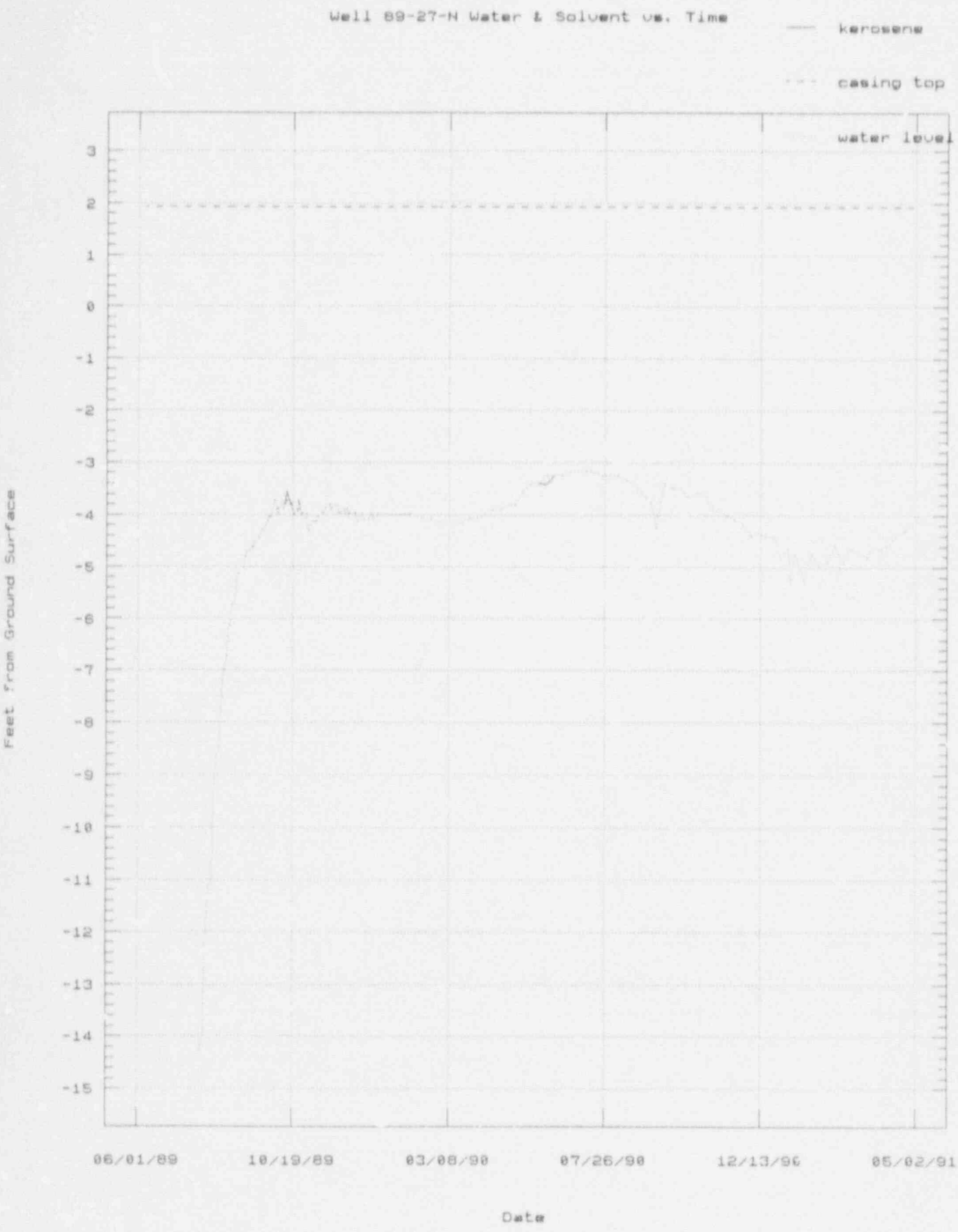
07/26/90
12/13/90
06/02/91
Date

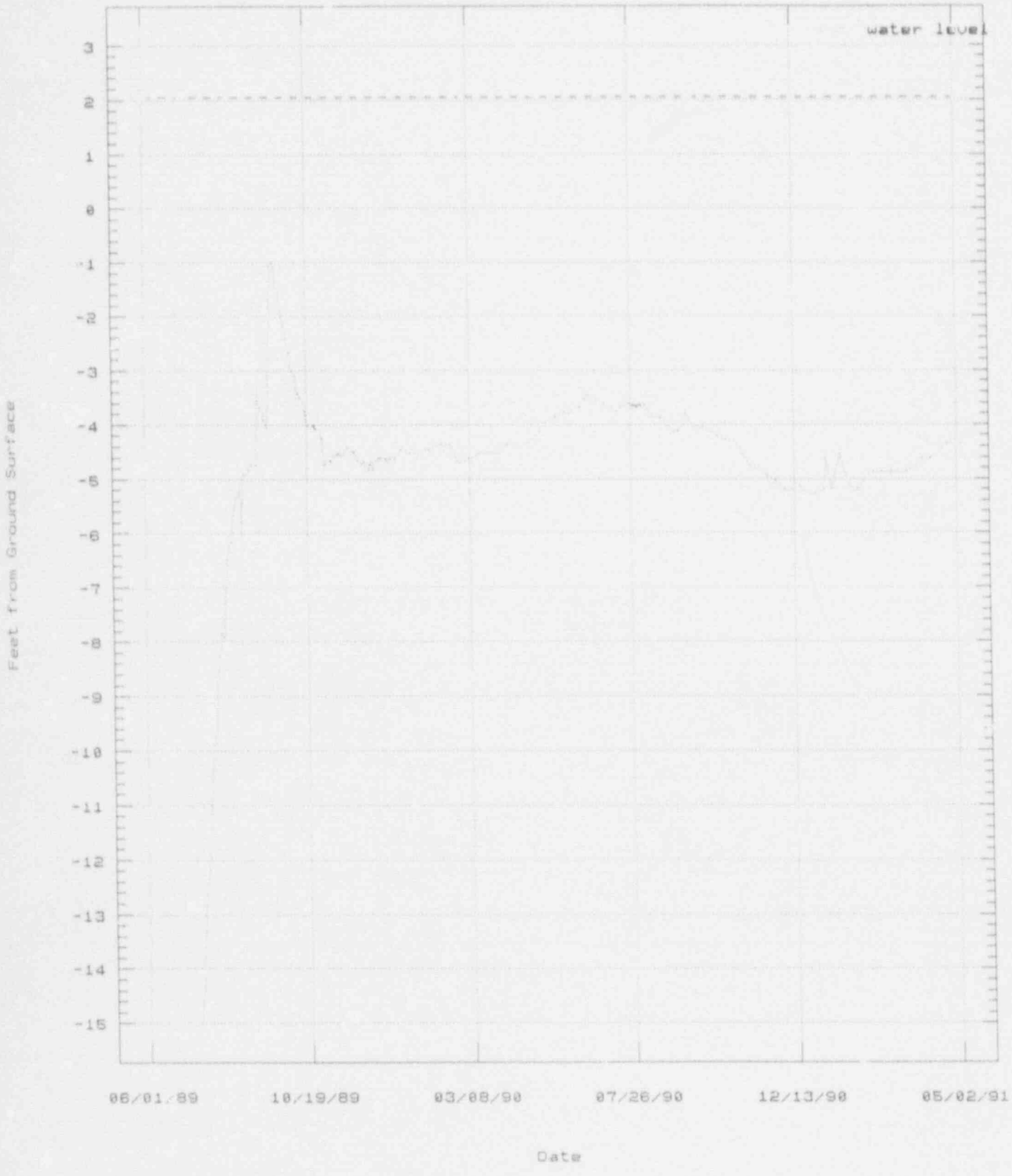

Wells 89-14-N & 89-14-E Water Levels
and Solvent Level vs. Time
14-N solvent
casing top
14-N water
14-E water
Feet from Ground Surface
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

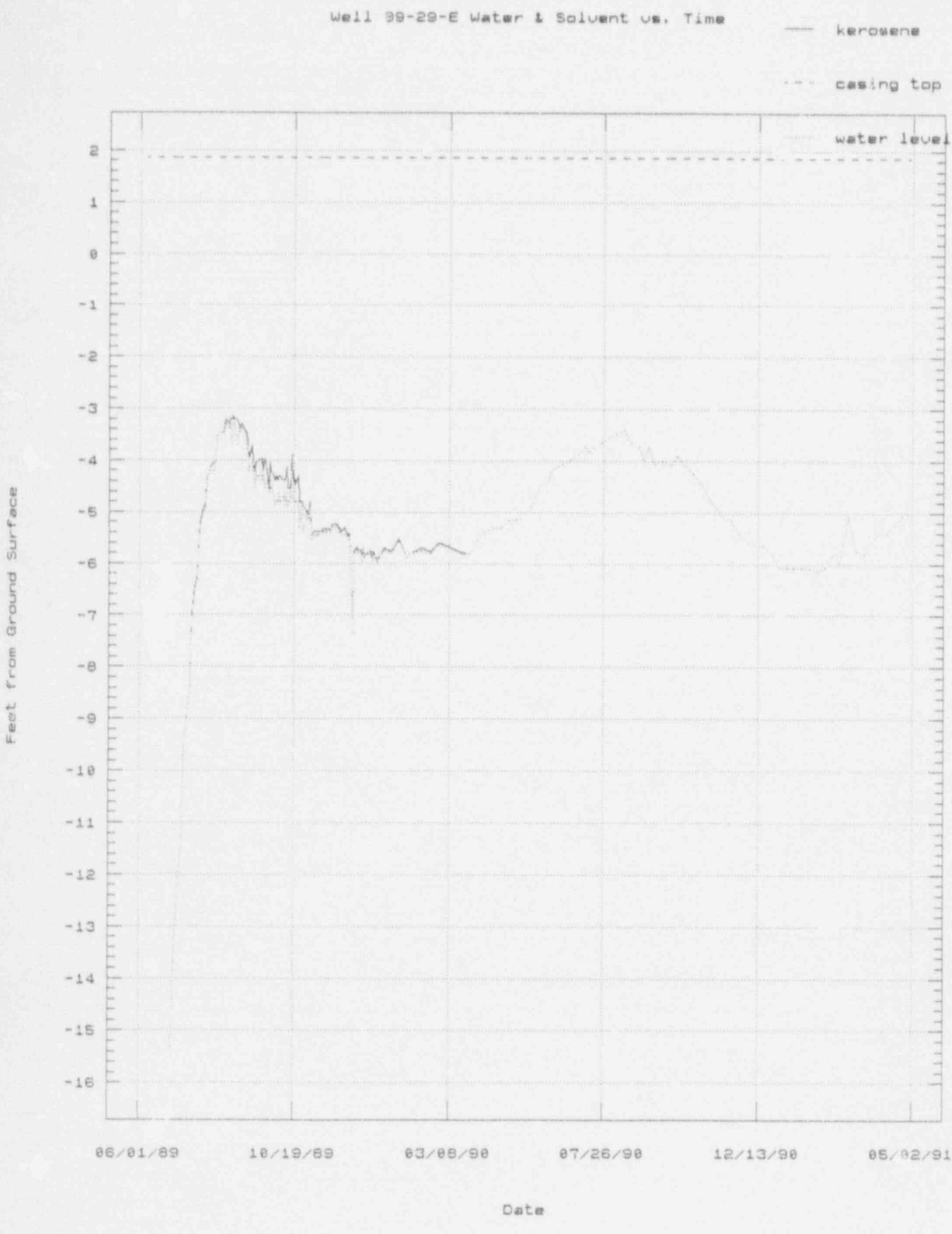

Wells 89-27-N & 89-27-W Water and
Solvent Levels vs. Time
27-N solvent
casing top
27-N water
27-W water
27-W solvent
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
-15
Feet from Ground Surface
06/01/89
10/19/89
03/06/90
07/26/90
12/13/90
05/02/91
Date

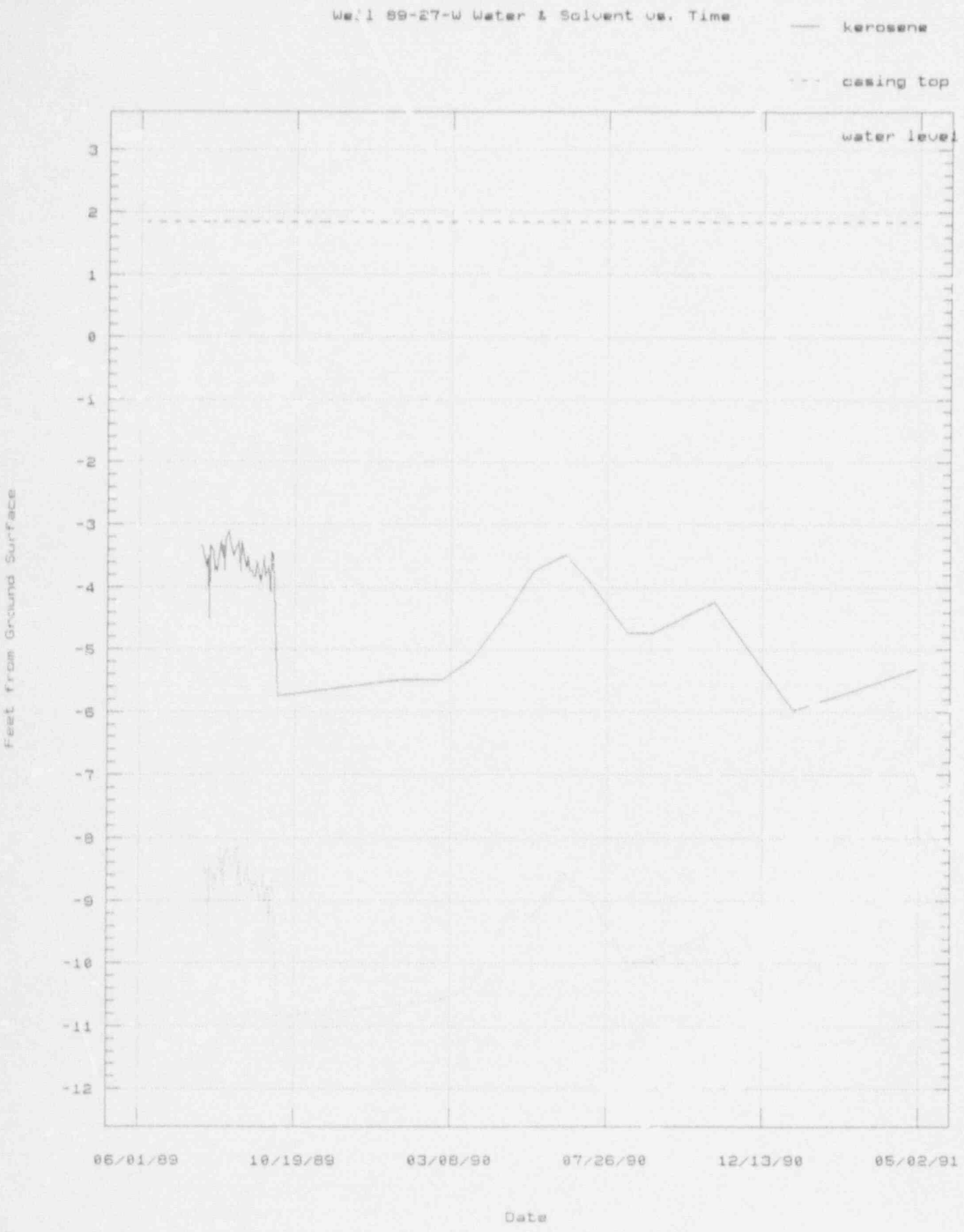

Wells 89-29-N & 89-29-E Water and
Solvent Levels vs. Time
29-N solvent
casing top
29-N water
29-E water
29-E solvent
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
-15
-16
Feet from Ground Surface
08/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

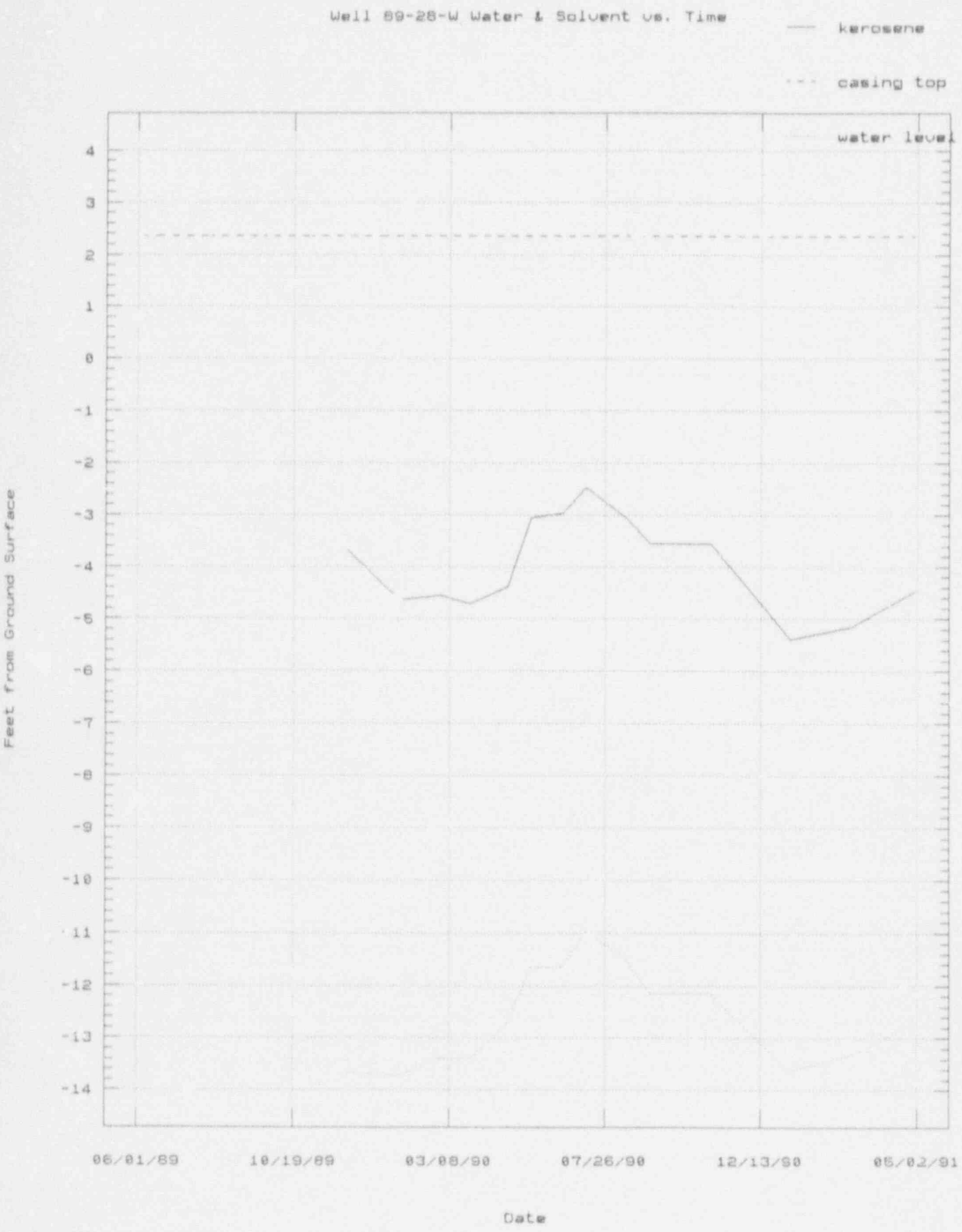

Well 89-14-N Water & Solvent vs. Time
kerosene
casing top
water level
3.4
2.4
1.4
0.4
-0.6
-1.6
-2.6
-3.6
-4.6
-5.6
-6.6
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

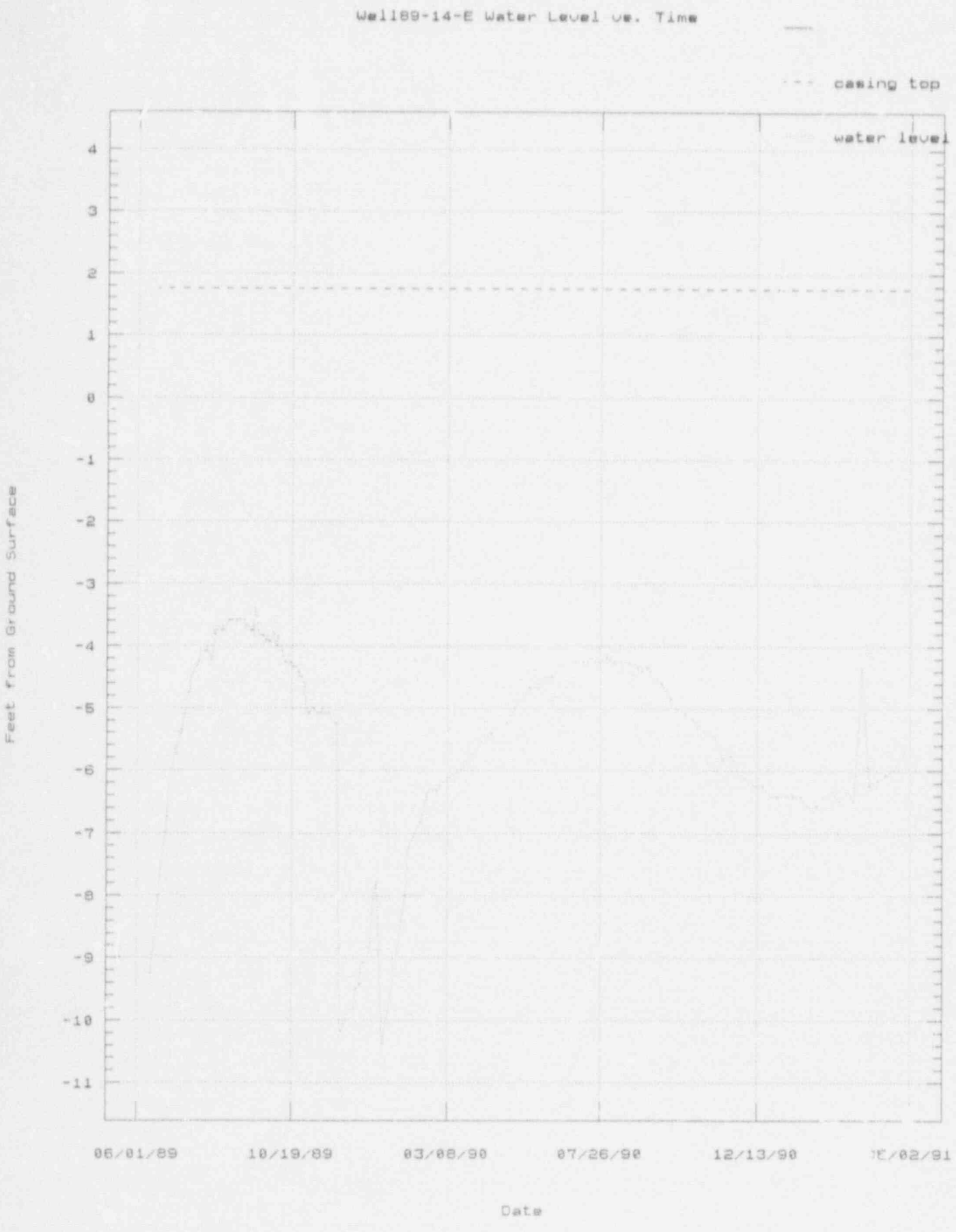

Well 89-27-N Water & Solvent vs. Time
kerosene
casing top
water level
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
-15
Feet From Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

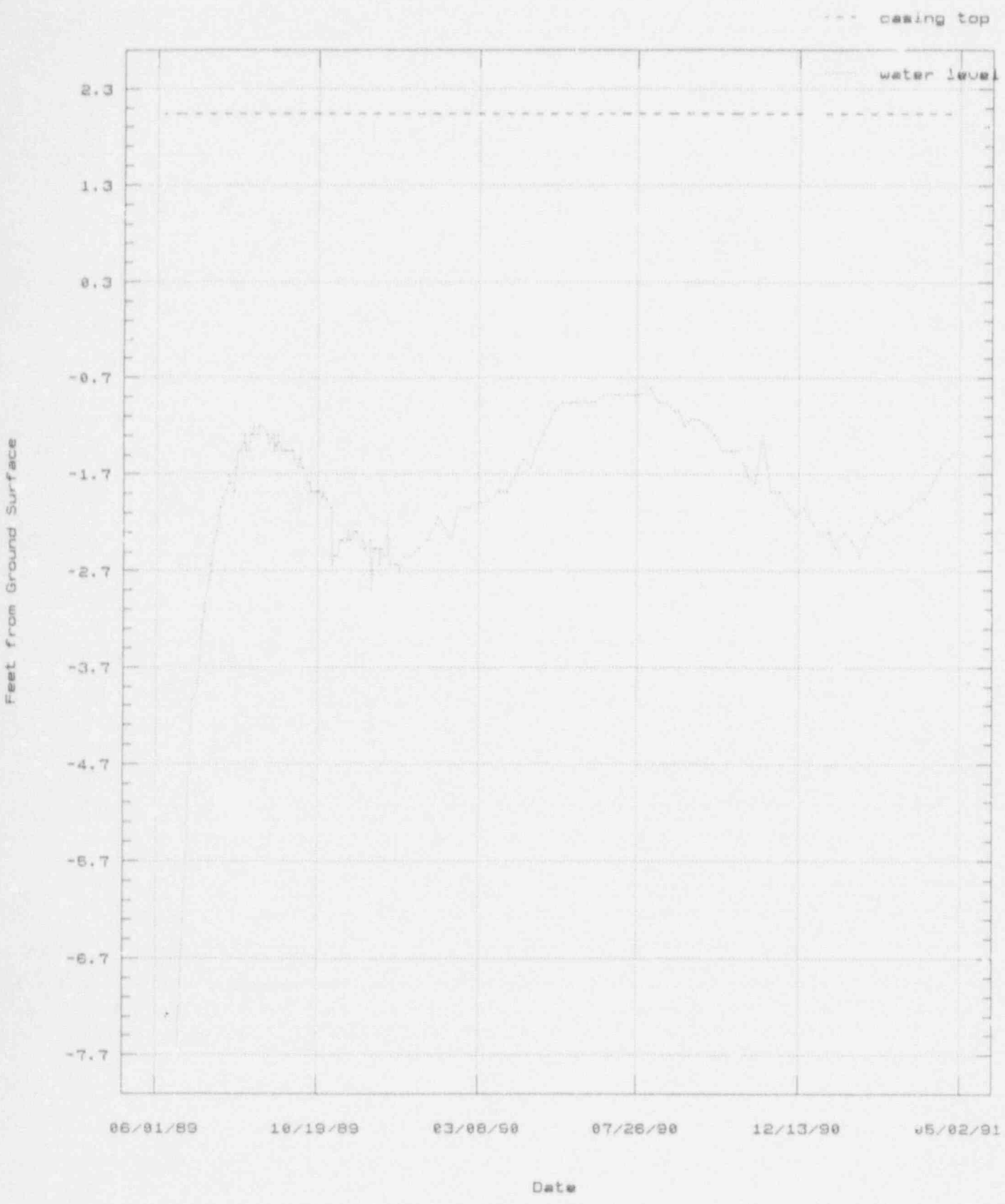

Well 89-29-N Water & Solvent vs. Time
kerosene
casing top
water level
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
-15
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

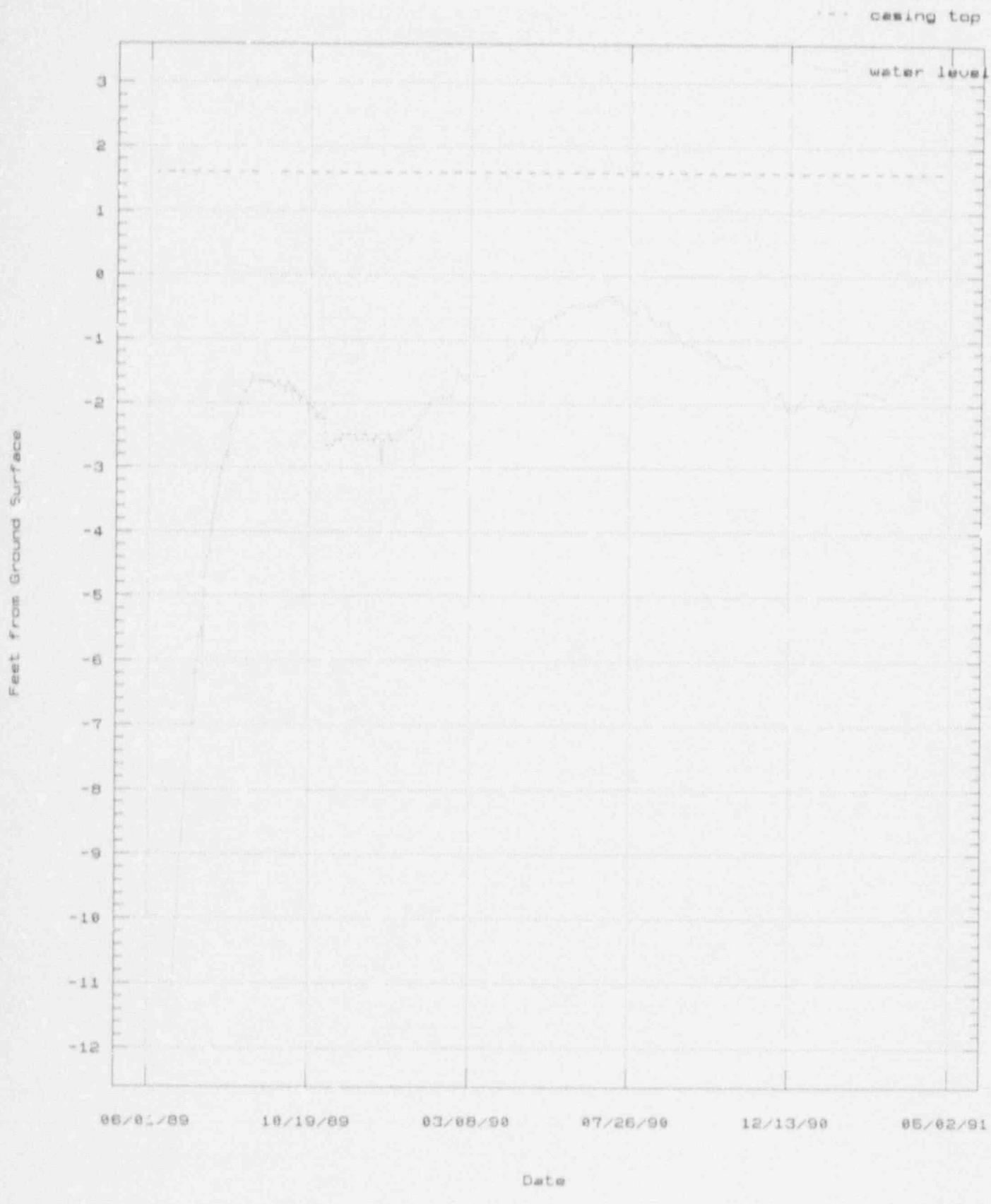

Well 99-29-E Water & Solvent vs. Time
kerosene
casing top
water level
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
-15
-16
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 89-27-W Water & Solvent vs. Time
kerosene
casing top
water level
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

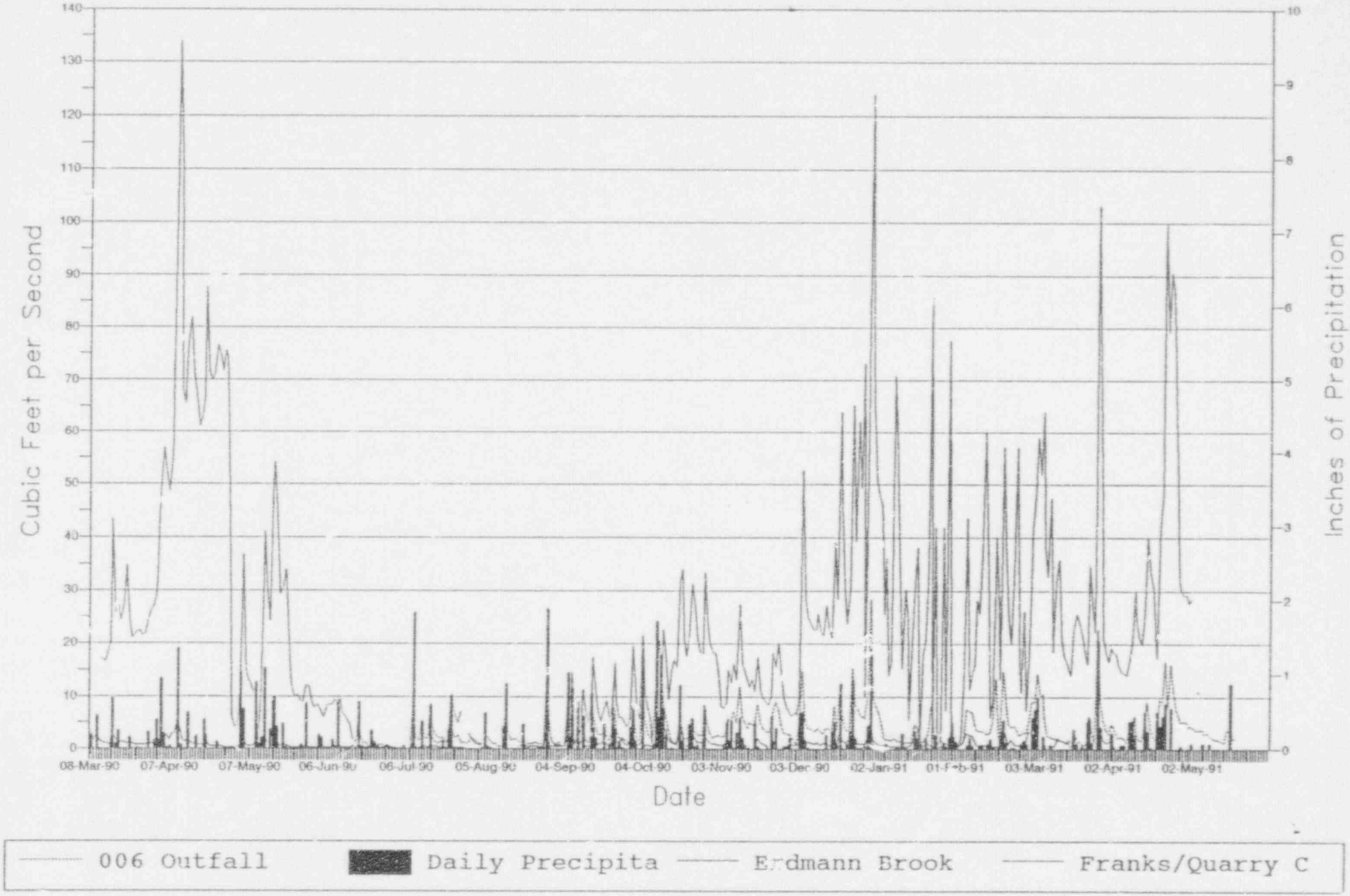

Well 89-28-W Water & Solvent vs. Time
kerosene
casing top
water level
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
-13
-14
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well89-14-E Water Level vs. Time
casing top
water level
4
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

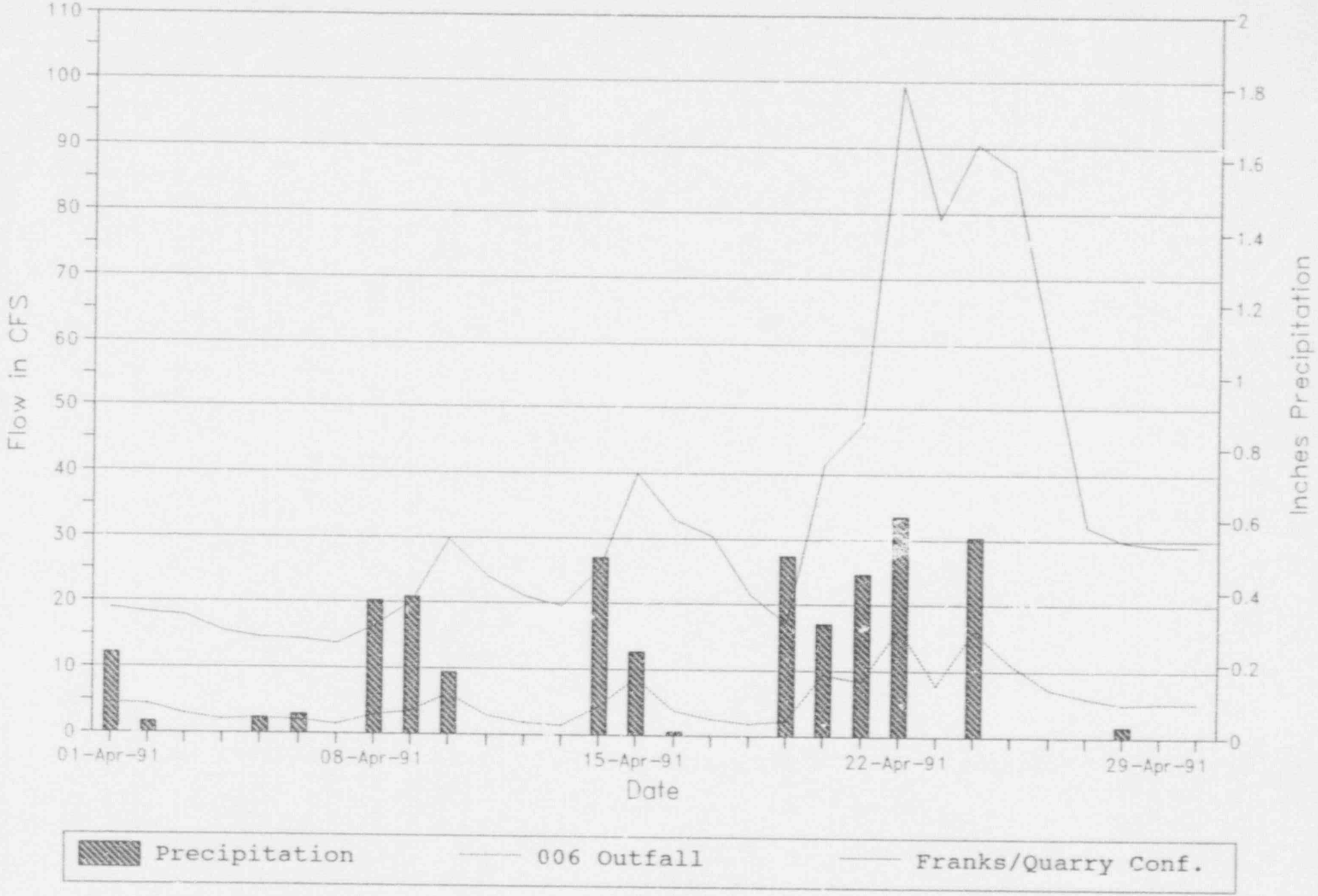

Well 09-13-E Water Level vs. Time
casing top
water level
2.3
1.3
0.3
-0.7
-1.7
-2.7
-3.7
-4.7
-5.7
-6.7
-7.7
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date


Well 89-13-W Water Level vs. Time
casing top
water level
3
2
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
-10
-11
-12
Feet from Ground Surface
06/01/89
10/19/89
03/08/90
07/26/90
12/13/90
05/02/91
Date

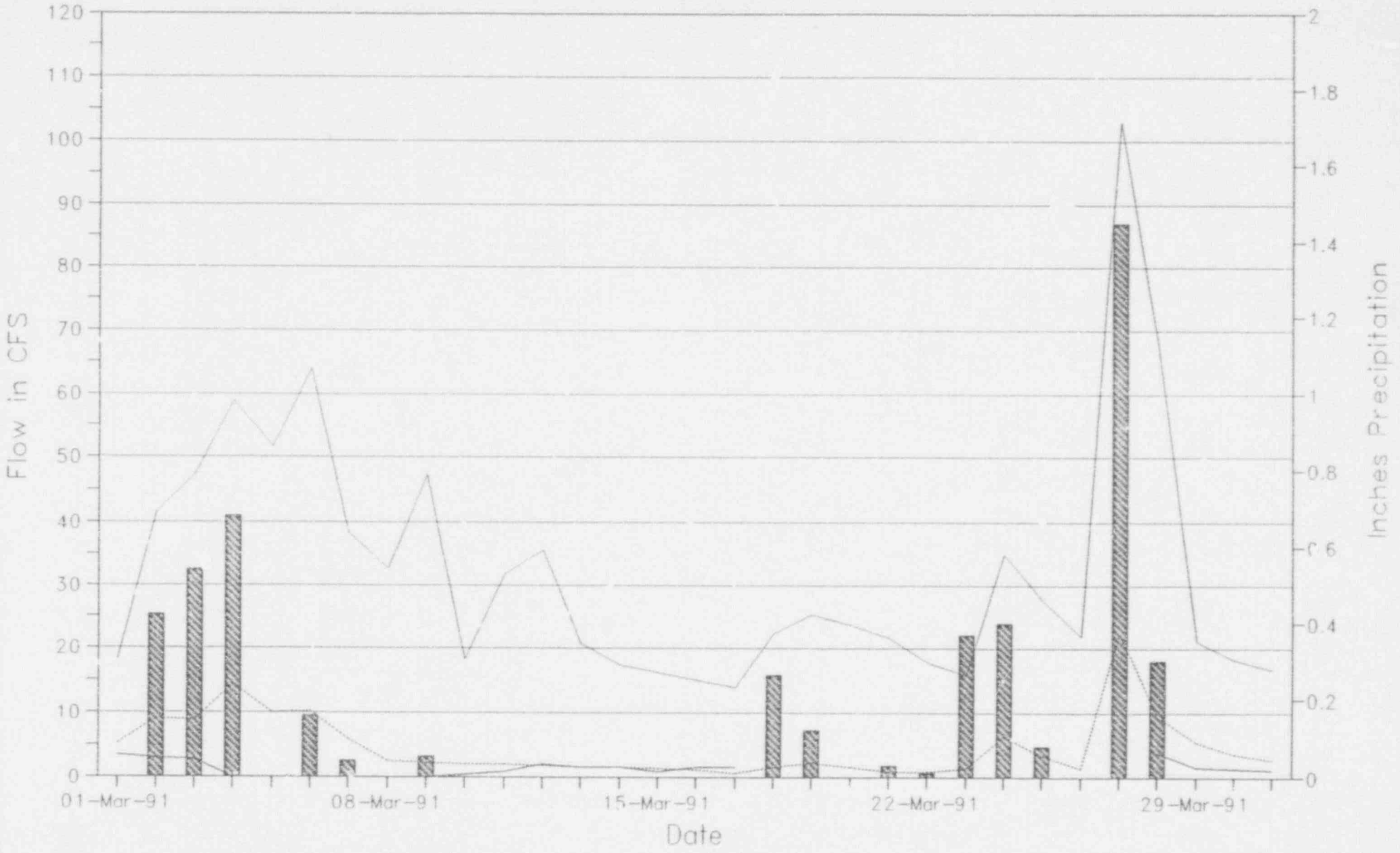
Erdmann Brook, 006 & Franks/Quarry Creek Flow and Daily Precipitation



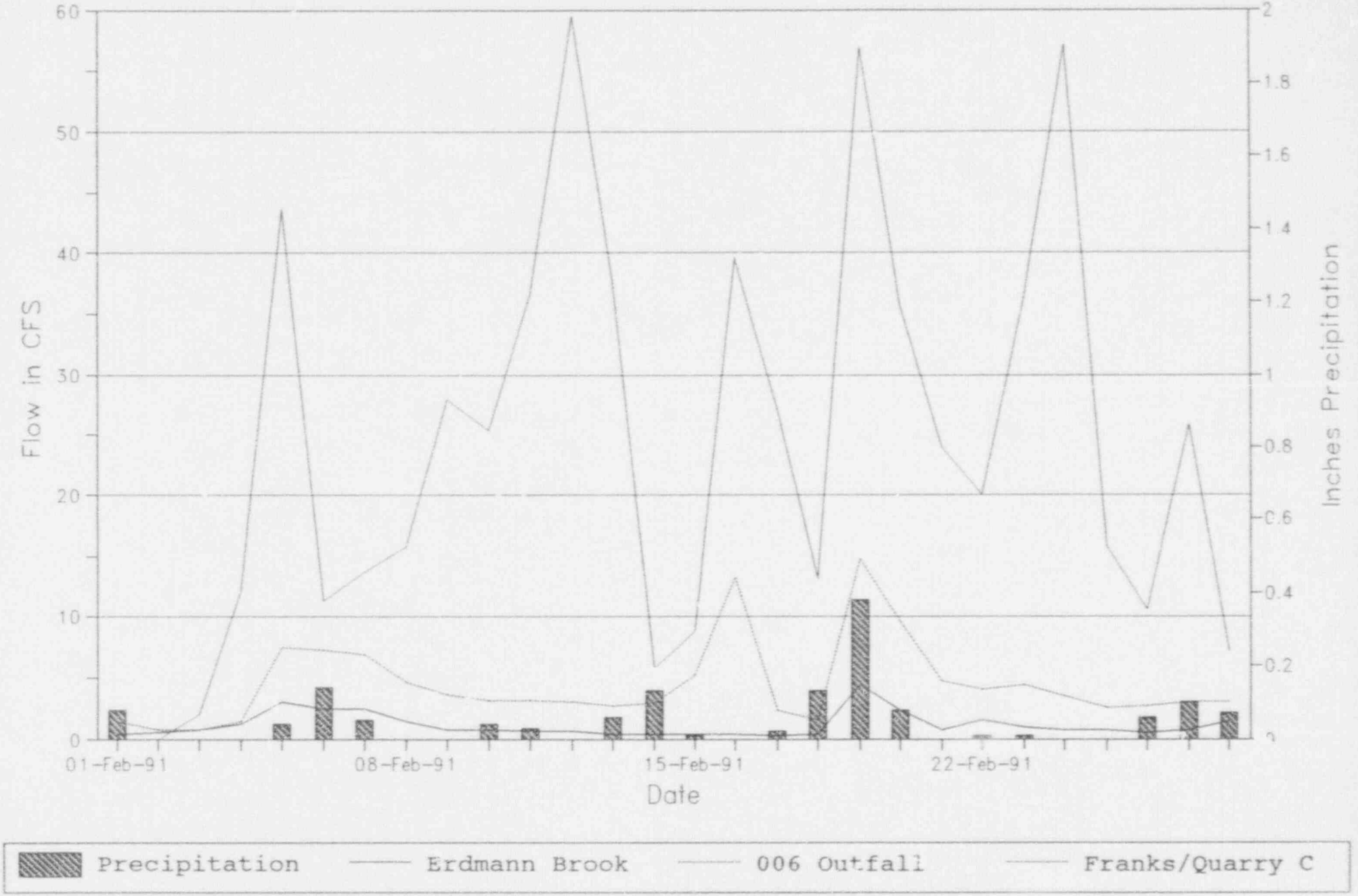
April 1991 Average Stream Flow and Daily Precipitation



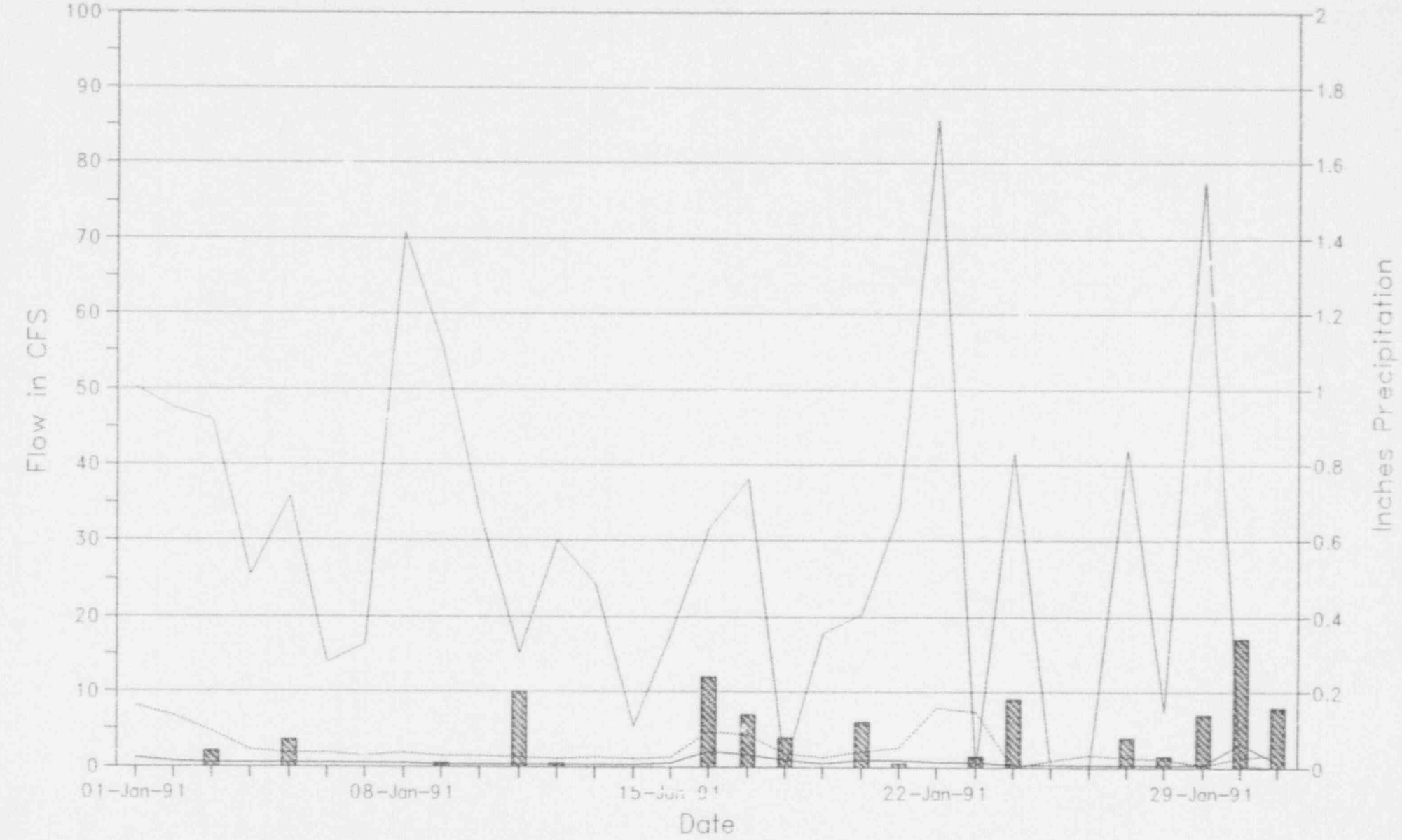
March 1991 Average Stream Flow and Daily Precipitation

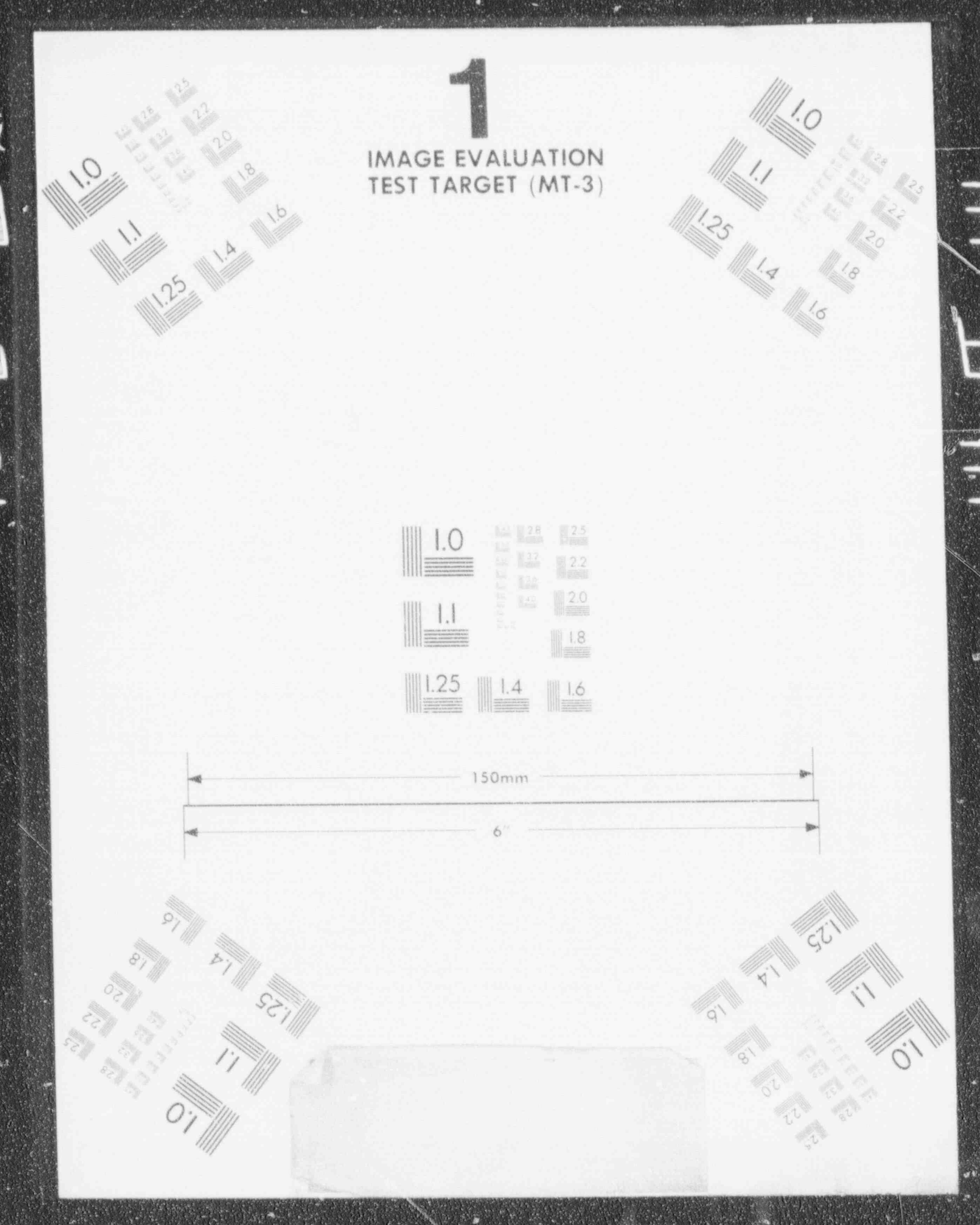


February 1991 Average Stream Flow and Daily Precipitation



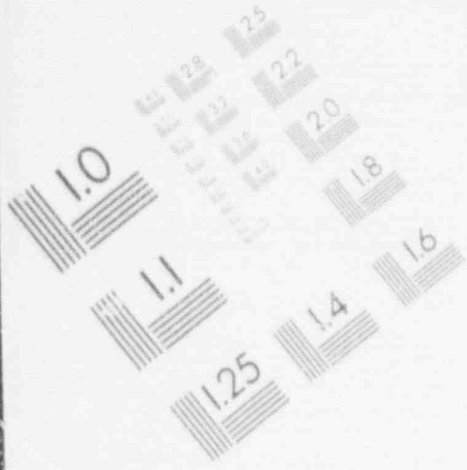
January 1991 Average Stream Flow and Daily Precipitation

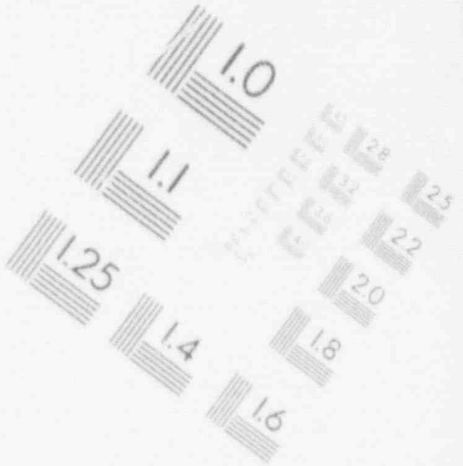


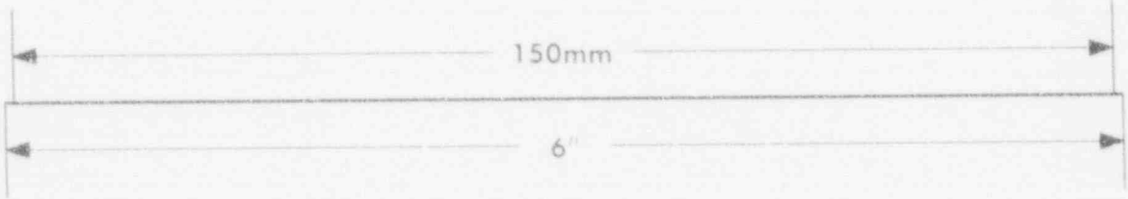

1
IMAGE EVALUATION
TEST TARGET (MT-3)
1.0
1.1
1.25
1.4
1.6
1.8
2.0
2.2
2.5
2.8
3.2
3.6
4.5
5.6
150mm
6"

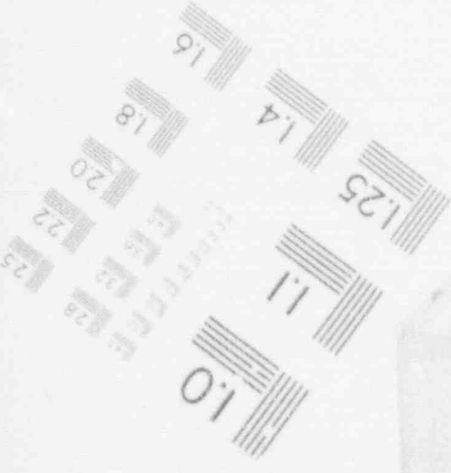
1

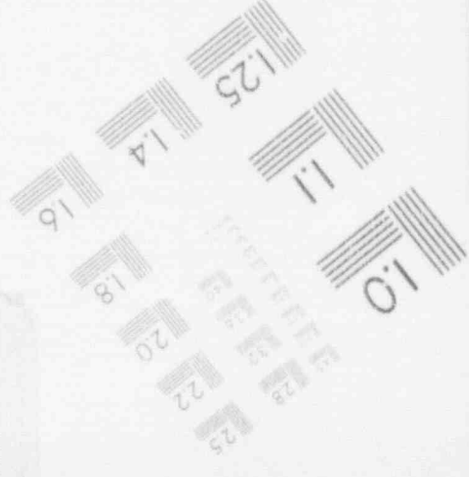
IMAGE EVALUATION TEST TARGET (MT-3)











1

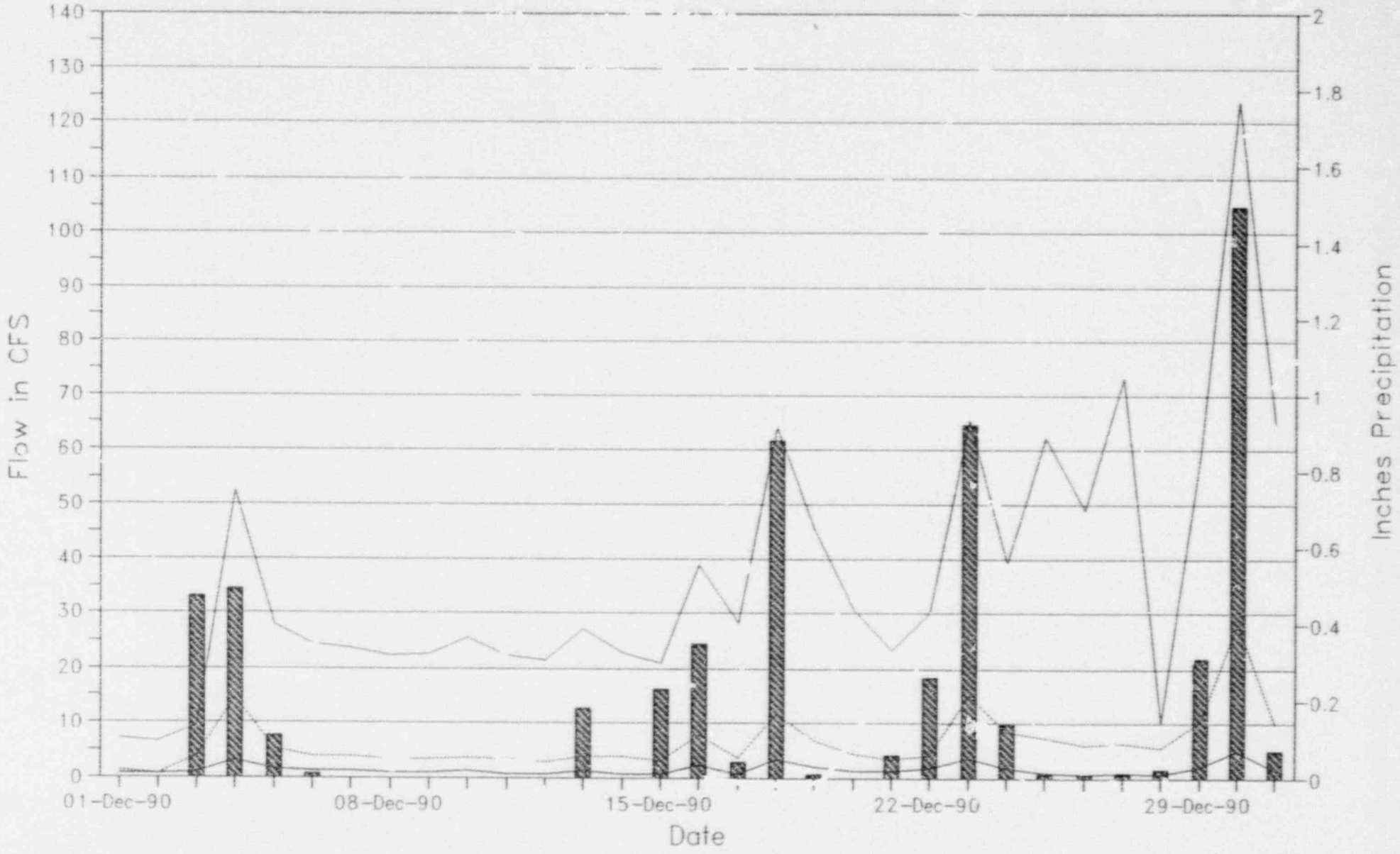
IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8

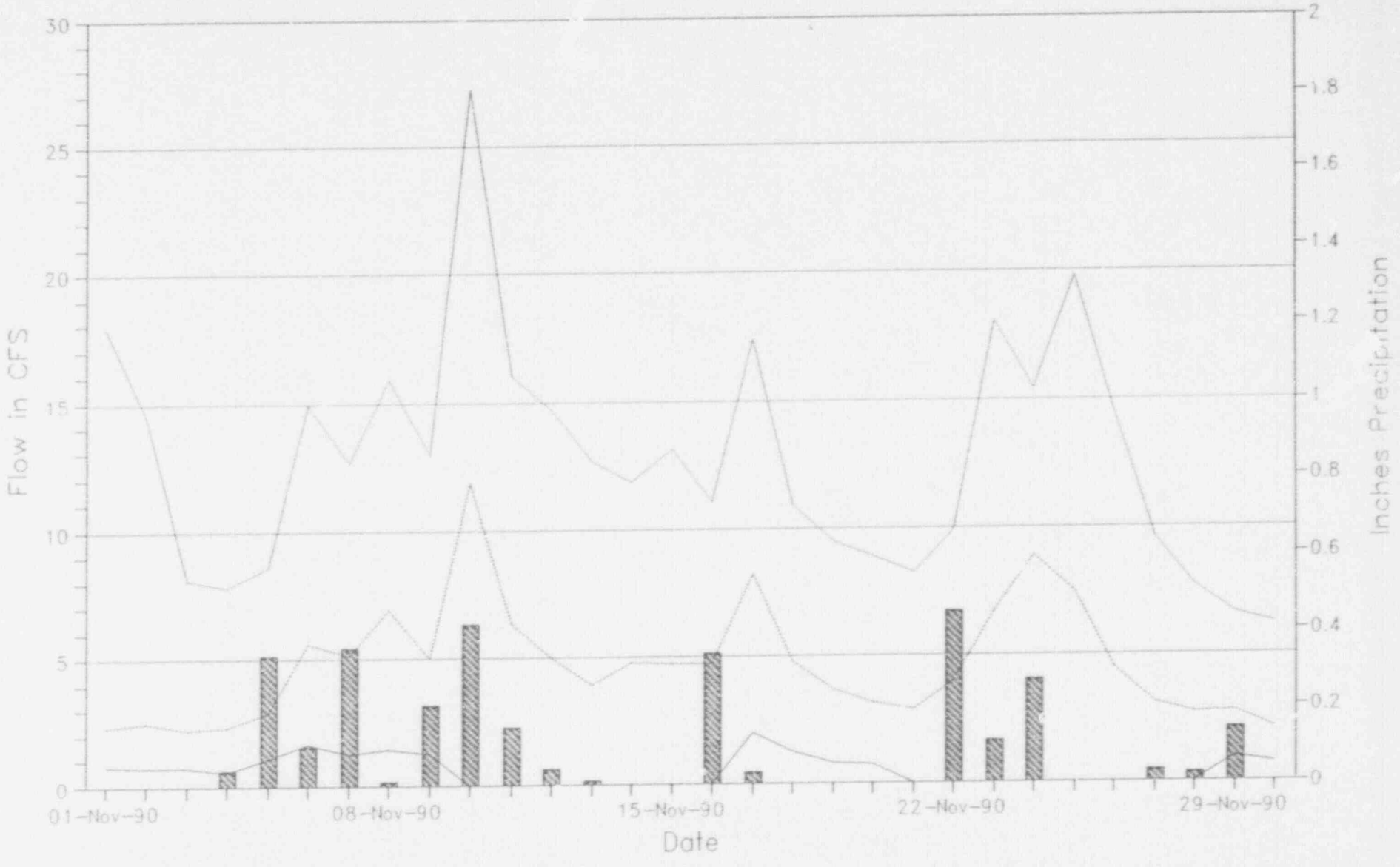
150mm

6"

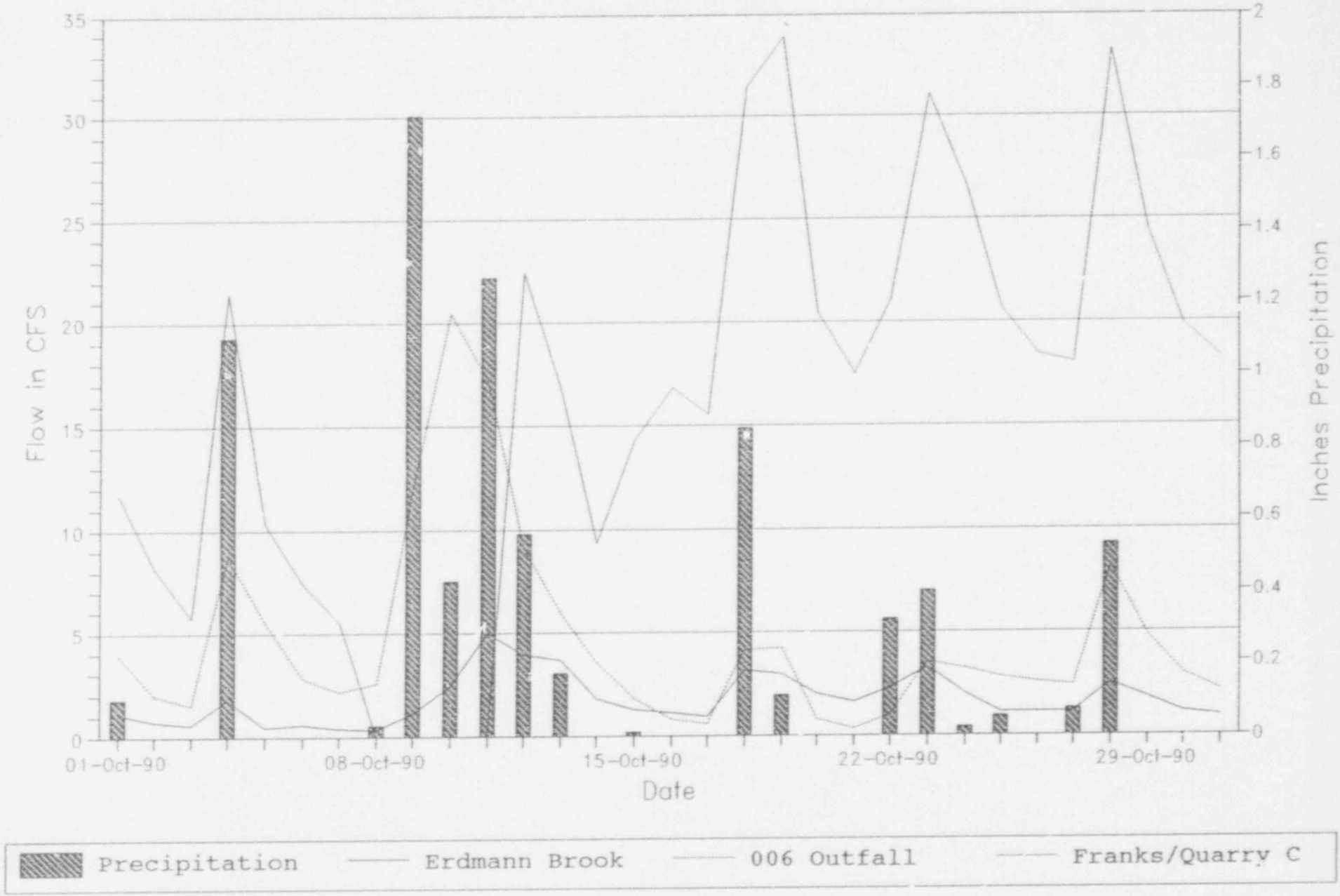
December 1990 Average Stream Flow and Daily Precipitation



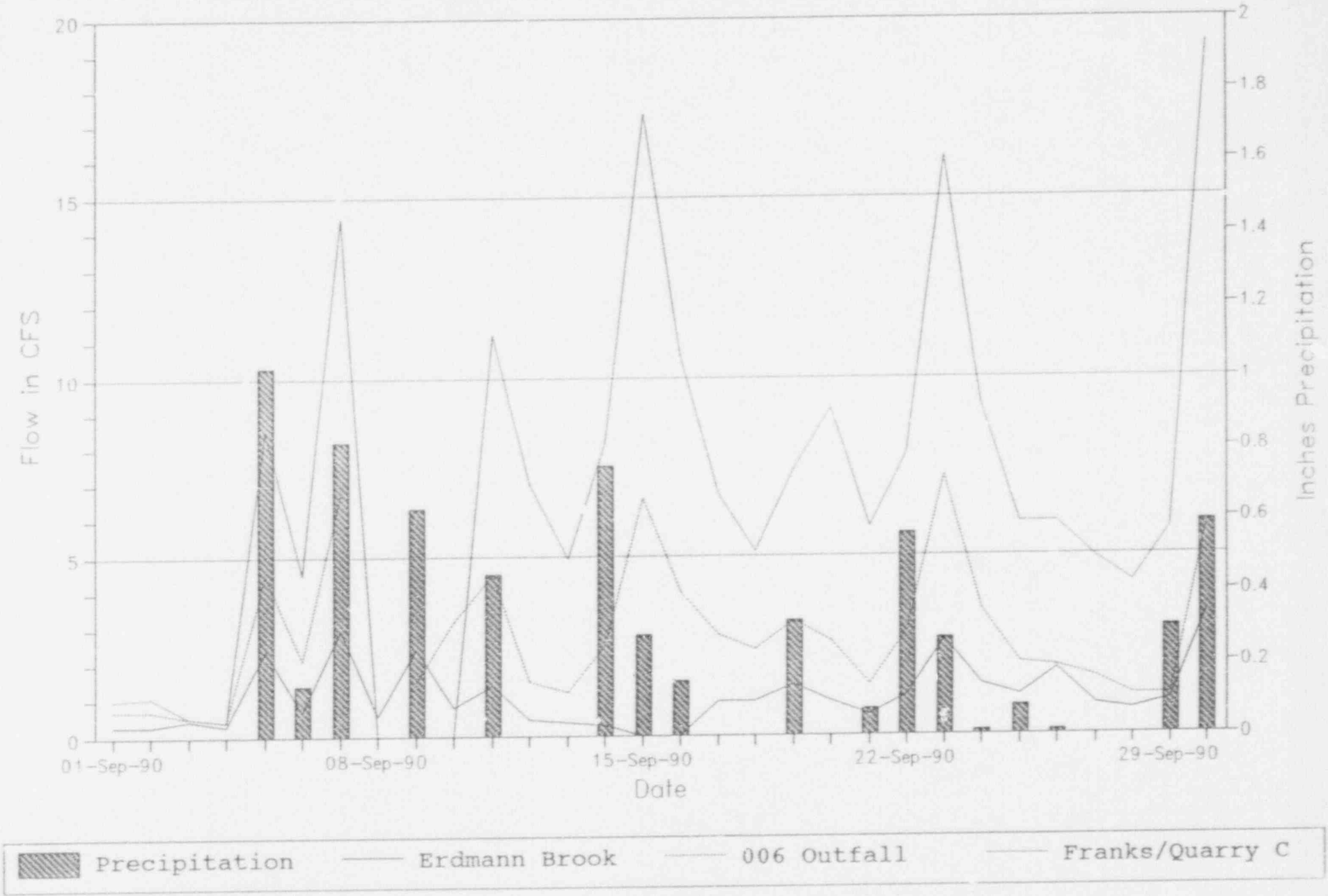
November 1990 Average Stream Flow and Daily Precipitation



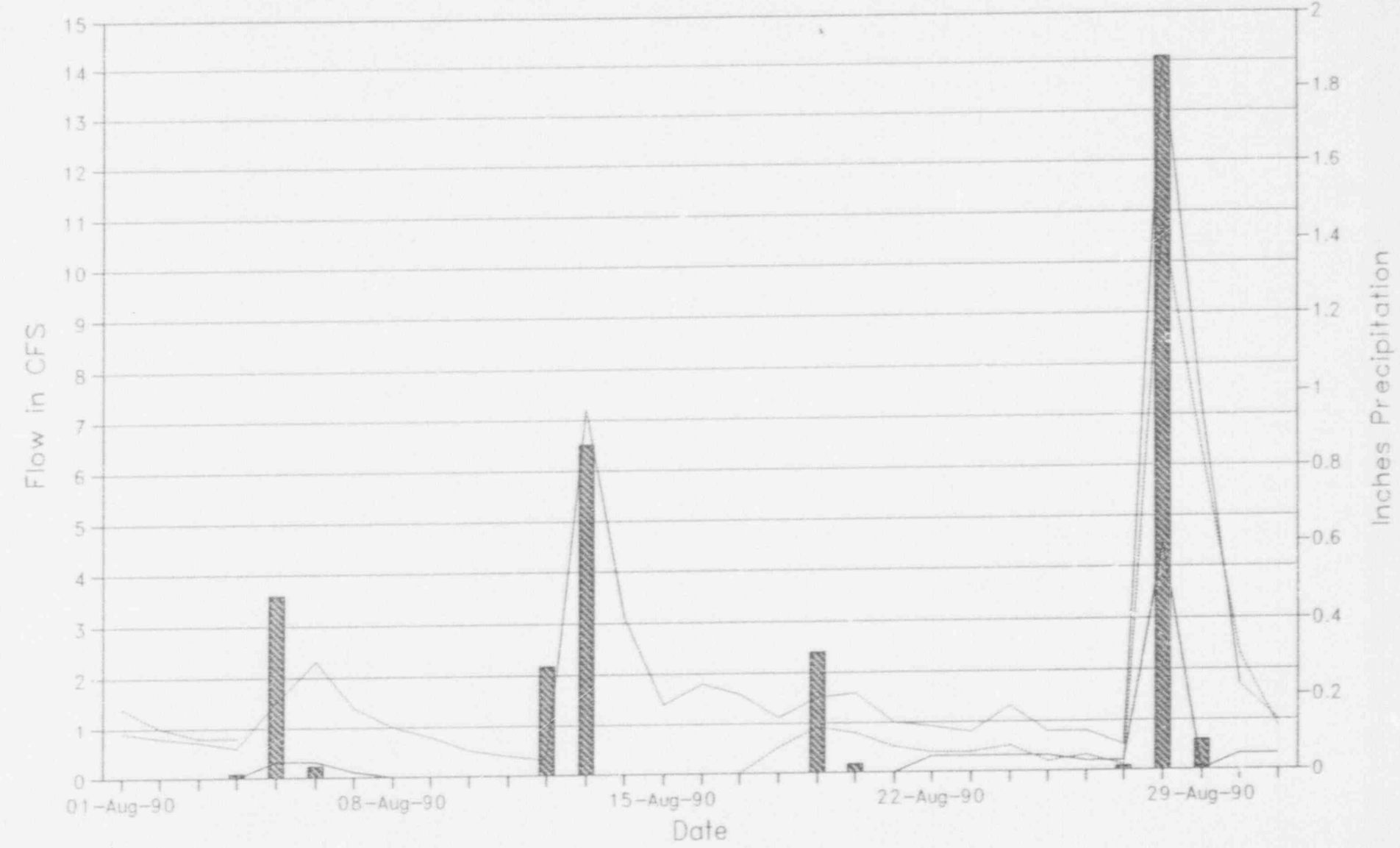
October 1990 Average Stream Flow and Daily Precipitation



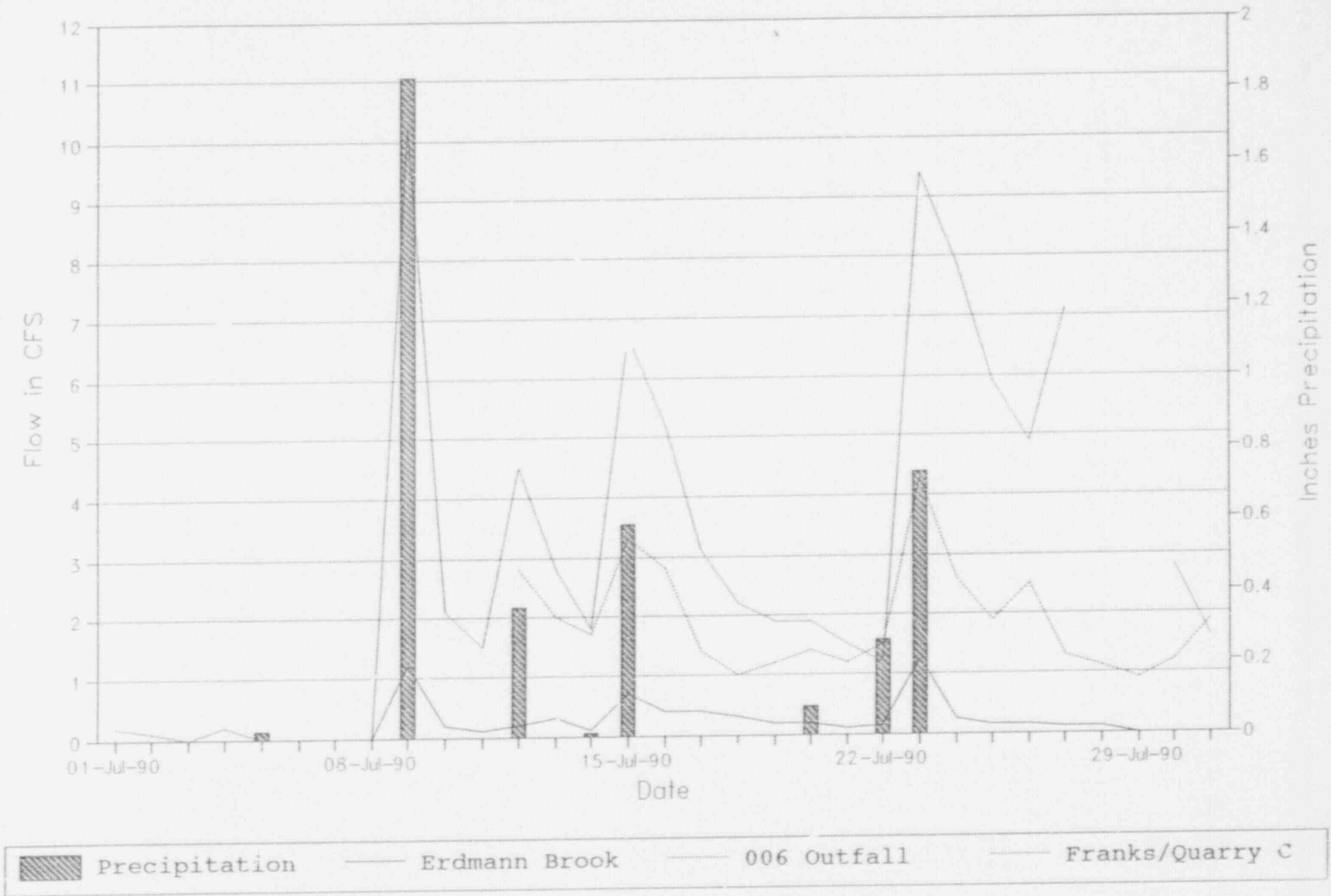
September 1990 Average Stream Flow and Daily Precipitation



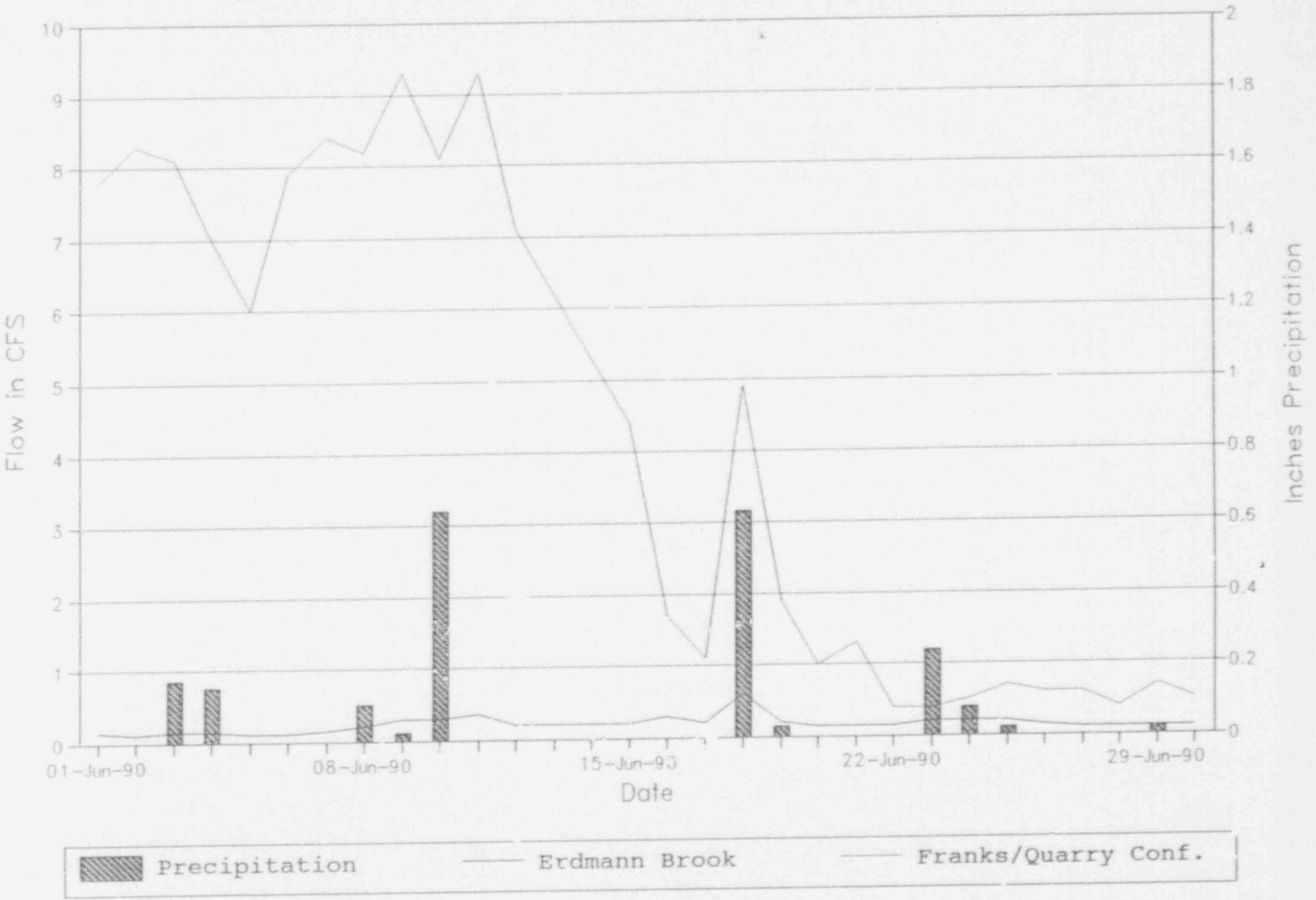
August 1990 Average Stream Flow and Daily Precipitation



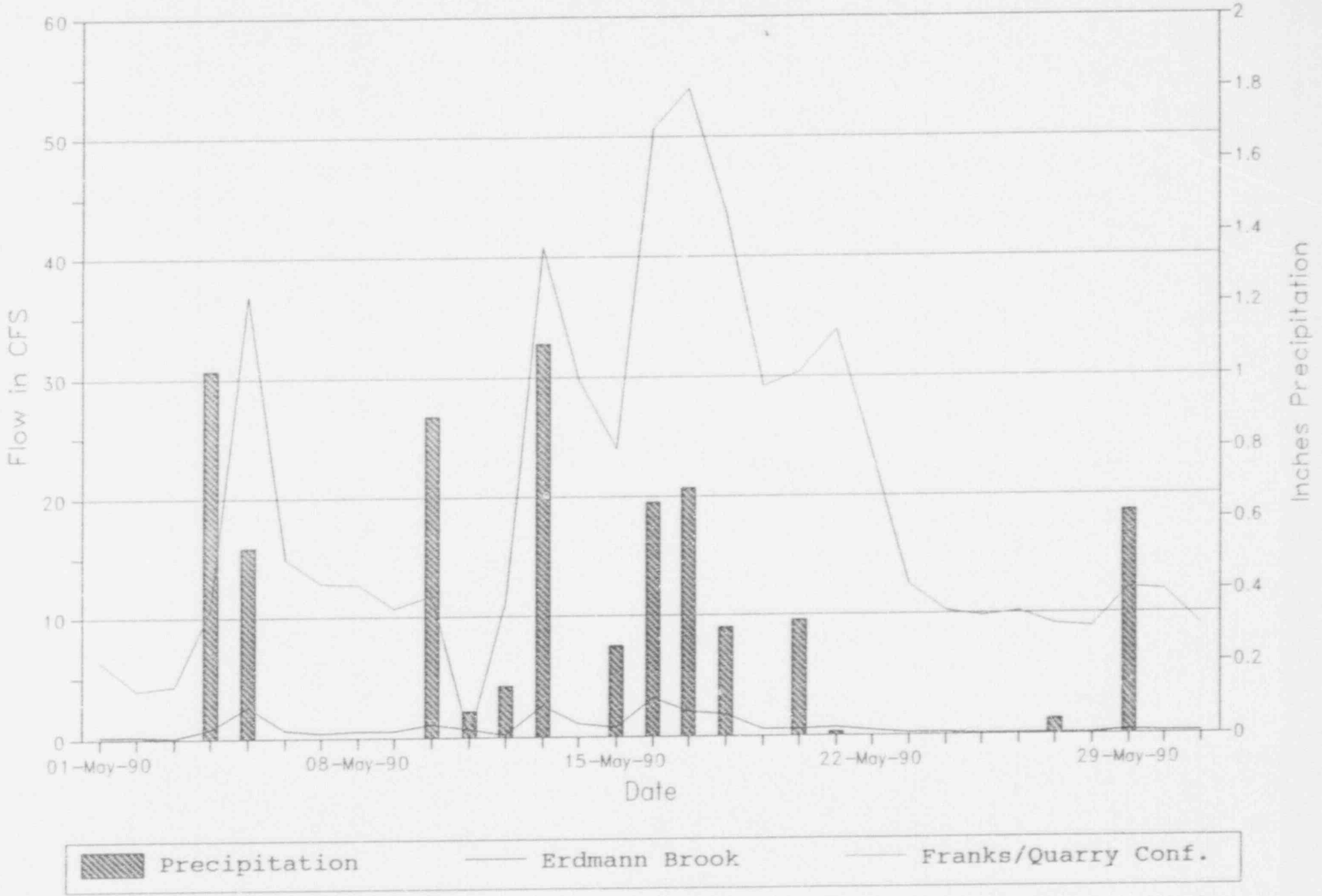
July 1990 Average Stream Flow and Daily Precipitation



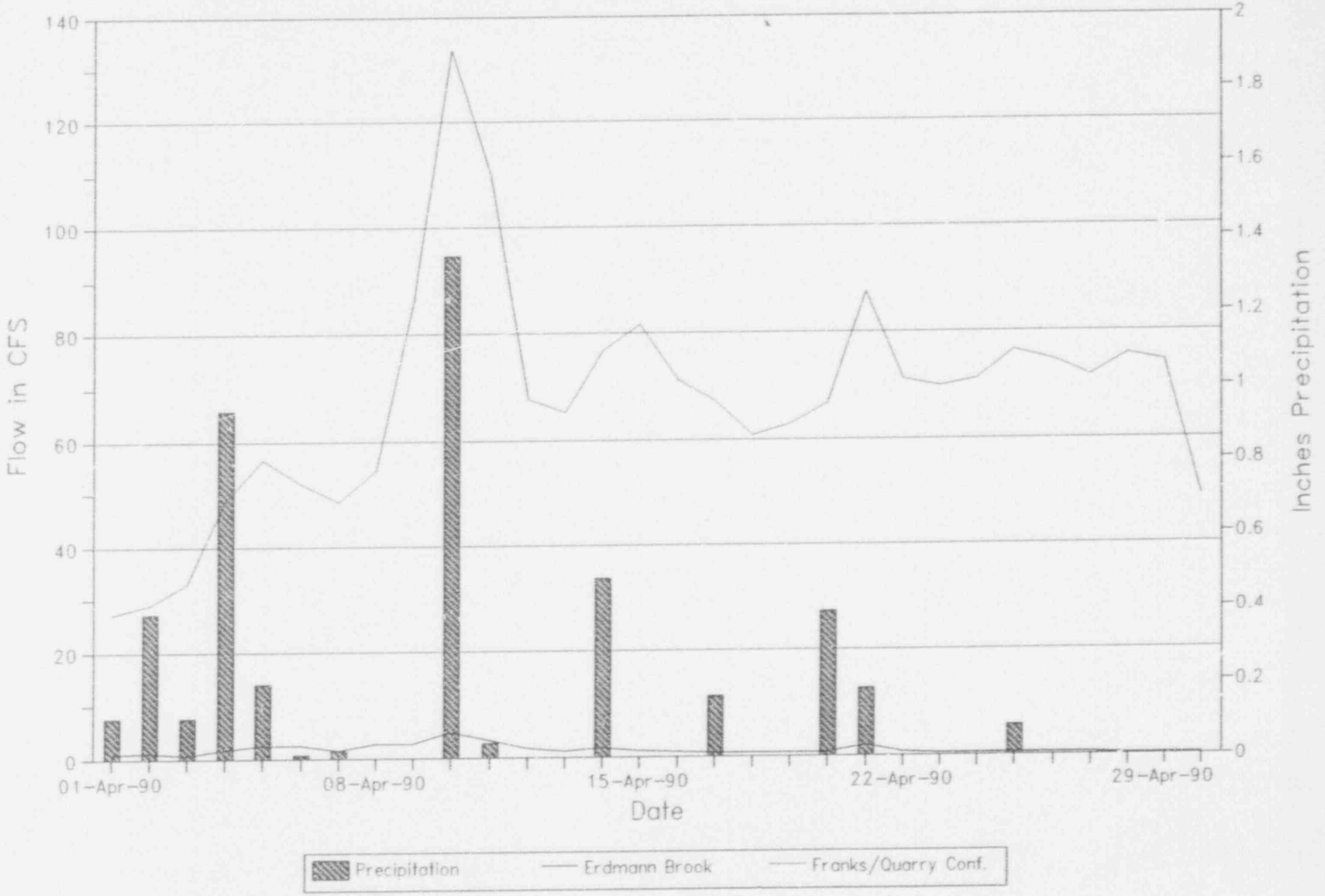
June 1990 Average Stream Flow and Daily Precipitation



May 1990 Average Stream Flow and Daily Precipitation



April 1990 Avg. Stream Flow and Daily Precipitation



March 1990 Average Flow Rates and Daily Precipitation